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The Environmental Protection Agency has review environmental impact statements for light-water reactors (LWR) since 1971. During the course of the reviews, EPA has emphasized the need for a thorough evaluation of the environmental risks, including risks from accidents, associated with LWR technology. The Reactor Safety Study is a comprehensive study of reactor safety and is the first such study to use a systems analysis approach in order to quantify the risks of reactor accidents in terms of probabilities and consequences, where historical and empirical data are inadequate. Since there are presently 53 LWRs licensed to operate and 188 more under construction, proposed or planned, it is imperative that the Reactor Safety Study be reviewed in depth and as impartially as possible so that the validity of the study's methodology and results can be determined.
This EPA report includes the two sets of formal Agency comments and the complete ITI report.

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REACTOR SAFETY STUDY (WASH-1400): A REVIEW OF THE DRAFT REPORT

August 1975

U.S. ENVIRONMENTAL PROTECTION AGENCY
OFFICE OF RADIATION PROGRAMS
WASHINGTON, D.C. 20460

EPA has

FOREWORD

The Environmental Protection Agency has reviewed environmental impact statements for light-water reactors (LWR) since 1971. During the course of our reviews, we have emphasized the need for a thorough evaluation of the environmental risks, including risks from accidents, associated with LWR technology. In August 1974, the Atomic Energy Commission (AEC) published for comment a draft report entitled "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants" (WASH-1400), which was the product of a major effort by the AEC spanning two years and costing 3 million dollars. The Reactor Safety Study is a comprehensive study of reactor safety and is the first such study to utilize a systems analysis approach in order to quantify the risks of reactor accidents in terms of probabilities and consequences, where historical and empirical data are inadequate. Since there are presently 53 LWRs licensed to operate and 188 more under construction, proposed or planned, it is imperative that the Reactor Safety Study be reviewed in depth and as impartially as possible so that the validity of the study's methodology and results can be determined.

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The Environmental Protection Agency (EPA) initiated its review of the Reactor Safety Study in August 1974, including an $88,000 contract to Intermountain Technologies, Inc. (ITI) of Idaho Falls, Idaho. The Office of Radiation Programs of EPA initially conducted a comprehensive review of the environmental consequence models, the general study methodologies and the conclusions, and issued formal comments to the Atomic Energy Commission on November 27, 1974. The ITI effort was directed at examining the study's evaluations of accidents and their event sequences to determine whether any significant failures of systems or equipment had been omitted or any major error or system biases of data analysis had been incorporated. For any areas identified with possibly significant errors or omissions, the impacts on the risks of the necessary adjustments in the variables or sequences of events were evaluated and estimated by ITI.

The final ITI report entitled "A Review of the Draft Report Reactor Safety Study (WASH-1400)" was submitted to EPA in July 1975. This report served as a partial basis for the preparation of the final Agency comments on draft WASH-1400, which were publicly issued on August 15, 1975.

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This report is now being published in its entirety so that it will be available as a resource to the scientific community and the general public. All Agency comments presented during the review are intended to provide constructive criticism which may be helpful to the Nuclear Regulatory Commission (NRC) as it prepares the final Reactor Safety Study report. (Upon the reorganization of the AEC to form the NRC and some portions of the Energy Research and Development Administration on January 19, 1975, the NRC assumed responsibility for completing the Reactor Safety Study.) This EPA report includes the two sets of formal Agency comments and the complete ITI report. When the final Reactor Safety Study is published by the NRC, EPA, with the assistance of ITI, will undertake a comprehensive review of that report and will publicly comment to the NRC.

We welcome comment on this report and would appreciate receiving any corrections or critical comments on the information and conclusions presented. Please send any such comments to the Environmental Protection Agency, Office of Radiation Programs (AW-559), Washington, D.C. 20460.

W. D. Rowe

W. D. Rowe, Ph.D.
Deputy Assistant Administrator
for Radiation Programs

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PART 1

NOVEMBER 27, 1974 COMMENTS TO THE AEC

UNITED STATES ENVIRONMENTAL PROTECTION AGENCY
WASHINGTON, D.C. 20460

27 NOV 1974

Mr. Saul Levine
Project Staff Director
Reactor Safety Study
U.S. Atomic Energy Commission
Washington, D.C. 20545

Dear Mr. Levine:

The Environmental Protection Agency's comments from the initial phase of its review of WASH-1400 ("An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants") are transmitted with this letter.

Because the assessment reported in WASH-1400 is expected to be a principal step toward establishing the accident risk associated with nuclear power plants, we are reviewing it in two phases. The first phase is represented by the enclosed preliminary comments based on a two-month review effort. The second phase will include an in-depth review of selected aspects of the study with technical assistance being provided to EPA through a contract with Intermountain Technologies, Inc. This second phase should be concluded by May 1975, at which time our intent is to issue a final report detailing all of our comments. During this period of continuing review we hope to maintain a close liaison with the Atomic Energy Commission so that our final report will reflect an up-to-date awareness of any resolution attained regarding comments by EPA or others on the draft report.

We have reviewed the work plan for our continuing effort with members of your staff as well as others in the technical community. We are also including it as a part of our review comments so that others may be cognizant of our planned efforts.

Our initial review indicates that the Reactor Safety Study provides an innovative forward step in risk assessment of nuclear power reactors. The general methodology and approach utilized in determining risk levels developed in the Reactor Safety Study appear to provide a meaningful basis for obtaining useful assessments of accident risks at nuclear power plants. Certainly, significant improvements in obtaining and utilizing nuclear plant operating data could considerably narrow the uncertainty range of risk estimates. We do, however, believe that certain aspects of the report require modification and information additions. In particular, the consequence modeling assumptions appear to underestimate the health effects resulting from the accident sequences associated with the larger releases of radioactivity. It is uncertain what the impact of this apparent underestimation may be on the resultant risk assessment.

Although the report does not make an absolute judgment on nuclear power plant accident risk acceptability, the comparative risk approach highlighted in the summary and the main volume of the study will certainly imply an acceptability judgment to the average reader. EPA recognizes that the comparative risk approach is a first step in addressing this question, but by itself is misleading. However, studies in progress by EPA and others indicate that judgments on "risk acceptability" are extremely complex, with comparative risk evaluations representing only one of numerous inputs which must be considered.

We are interested in the plans for application of this methodology to other reactor systems and other components of the nuclear fuel cycle. Certainly, we would recommend that studies of this type should be considered by the applicable AEC successor and that their intent in these areas be publicly stated.

We would be pleased to discuss our comments with you if they require any clarification.

Sincerely yours,

W. D. Rowe

W. D. Rowe, Ph.D.
Deputy Assistant Administrator
for Radiation Programs (AW-558)

Enclosure

ENVIRONMENTAL PROTECTION AGENCY

WASHINGTON, D. C. 20460

November 1974

Comments by the Environmental Protection Agency

on

Reactor Safety Study

AN ASSESSMENT OF ACCIDENT RISKS IN U.S. COMMERCIAL NUCLEAR POWER PLANTS

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INTRODUCTION AND CONCLUSIONS

Review Perspective

The Environmental Protection Agency has completed a preliminary review of the draft report "Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400, prepared by the Atomic Energy Commission. Our review process will continue through April 1975 at which time we will issue a final set of comments. During this period of continuing review, we hope to maintain a close liaison with those responsible for the Study so our final comments will reflect an up-to-date awareness of any resolution attained regarding comments by EPA or others on the draft report.

EPA's review of the Study cannot be considered as exhaustive in that many of the calculational details and data base have not been checked. Our focus has rather been one of emphasizing a review of major assumptions, concepts, methodology and approach. Although EPA's resources are limited when compared with those utilized in the development of the Study, it was deemed necessary to do as comprehensive a review as possible due to the many significant implications the Study has with regard to areas of EPA responsibility. EPA's primary concerns deal with the health and safety of the public and the protection of the environment from the consequences of accident releases. In this respect, we are involved in the planning for mitigation of these potential consequences, including the development of appropriate Federal guidance, and in assuring that the public risks incurred are societally acceptable. Within this context we attempt to maintain cognizance of accident analysis activities so that we can be continually aware of both the probability of accidents and the consequences for such accidental releases. Due to the importance we attach to this subject and the broad range of subject matter considered in the Study, we believe it is imperative that it receive a thorough and critical review by the general technical community and the public. We realize that much of this review that we suggest is already underway or planned. However, we feel that comments developed on the Study should be referenced in the final version of the Study and copies of these reviews should be publicly available.

In continuing its review, EPA has contracted with Intermountain Technologies, Inc. to assist us in the evaluation of the range of applicability of the various analytical models and assumptions utilized in the assessment. The preliminary work plan for this effort is presented in an attachment to our comments. If, in the initial stages this detailed review of the selected failure mode paths or critical source term parameters indicates a general agreement with the Study's evaluation, that portion of the investigation will be terminated and other failure mode paths or source term parameters may be substituted in this work plan.

Review Format

Following this Introduction and Conclusions section, our review takes up individual groups of volumes of the WASH-1400 document by first presenting general comments and then specific comments. This sequence begins with the main volume of the Study and continues with Appendix VI (environmental consequences); Appendices V, VII, and VIII (accident sequence, meltdown processes, and radioactivity releases); Appendices I, II, III, and IV, (definitions of failure pathways), Appendix X (design adequacy) and the summary volume, in that order. The last section of our review presents Additional Comments in order of the Study volumes themselves. These latter comments were not felt to be of the same level of significance as those referred to in the previous sections of our review.

Main Comments and Conclusions

EPA has made a broad spectrum of specific comments on the Study, realizing that they have varying degrees of impact on final results. However, as the document is bound to be used as a reference for many follow-on studies and analyses, we feel it is desirable to make it as complete and accurate as possible in all its facets. EPA's main comments and conclusions, although of a preliminary nature, are as follows:

1. The Study is innovative in both its concept and methodology and provides an innovative forward step in risk assessment of nuclear power reactors. In this respect, the AEC is to be commended. The general methodologies and rationale developed in the Study to determine risk levels appear to provide a meaningful basis for obtaining useful assessments of accident risks of nuclear power plants.

2. Appendix VI (environmental consequences) received particular attention in our review due to its pertinence to EPA concerns. This appendix was found to be quite weak in a number of respects and not up to the general thoroughness that appears to permeate many other sections. Our preliminary review indicates, for example, that if the recommendations of the BEIR Report are followed, the consequences estimated in the Study may be low, in certain cases, by factors of 2 to 5. In addition, the evacuation model assumed for the reference case consequence calculation also appears somewhat overly optimistic. Based on the information presented in the Study, this could increase consequences by at most a factor of 2 to 4 (i.e., no evacuation). Therefore, the combination of these factors could result in an underestimate, by about an order of magnitude, of the consequences associated with the "high" release accident sequences. Since the high release accident sequences are significant, but not dominating, contributors to the overall risk assessment, the resultant assessed

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risk magnitude would be increased but by a lesser factor. It is suggested that appropriate modifications should be made or the rationale for utilizing other assumptions should be provided.

Furthermore, the description of certain critical portions of the overall calculational process should be significantly expanded to permit a clear understanding of the relationships between the radioactive material releases, its dispersion, population distributions, and the resulting health effects.

3. Although the Study indicates that no absolute judgment on nuclear power plant acceptability is intended, the comparative risk approach highlighted in the summary may well imply an acceptability judgment to the average reader. It should be further pointed out in the report that the comparative risk approach is only a first step in addressing this question and by itself can be misleading. It can be noted that studies in progress by EPA, National Science Foundation, and others, indicate that judgments on "risk acceptability" are extremely complex, with comparative risk evaluations representing only one of numerous inputs which must be considered.

4. As can be expected with such a voluminous report, a number of apparent inconsistencies, format difficulties, and cases of insufficient supporting information were encountered. Particularly in Appendix II there were inconsistencies in identification of components and levels of detail in the various fault trees and system descriptions. There were also problems with the lack of a readily accessible glossary of abbreviations and with inadequate cross-referencing among appendices. It is suggested that the Study be subjected to the necessary editing to eliminate abbreviations wherever possible, glossaries be added for those abbreviations used (e.g., foldout in Appendix I) and the cross-referencing between appendices be improved. The formats employed in Appendices III and VII are worthy of consideration for use in all appendices.

5. There is some concern relative to a lack of certainty as to what the follow-on actions in this program area will be. This is intensified by the recent reorganization of the AEC and its functions and lack of definition as to where this effort will be picked up and continued. We would expect that some follow-on effort should be directed toward additional verification that the design, operational, or other variations among the 100 nuclear plants to which the Study is applied, do not significantly affect the overall risk calculated by the Study. Of major interest would be consideration of other plant designs such as Westinghouse 2 and 4 loop reactor coolant systems, and BWR Mark II and III containment designs. Other items such as the use of hydrogen recombiners and differing modes of containment spray injection should also be considered for

examination. It is realized that in many of these cases what may appear as a significant difference on the system or component level may not significantly change the overall risks but some documentation of this should be presented in order to show that to be the case. A further concern relative to continuing effort in this area but not related to this specific Study is the application of this methodology to other reactor systems and other components of the nuclear fuel cycle. Certainly we would recommend that these studies should be considered by the applicable AEC successor and that their intent in these areas be publicly stated.

ASSESSMENT OF ACCIDENT RISKS IN U.S. COMMERCIAL NUCLEAR POWER PLANTS

MAIN VOLUME

General Comments

The main volume presents a well written introduction to and summary of various analyses presented in the supporting appendices; therefore, comments on the material within this volume are generally covered elsewhere in this review. The discussion in Section 5.3 pertaining to the process of assessing release category probabilities was especially informative. The assignment to a category release probability of a 10% contribution from adjacent categories certainly adds a significant element of conservatism to the resulting probability values. The Study also attributes additional conservatism to the Monte Carlo process used to assess failure rate median values. However, the degree of conservatism attached to the Monte Carlo process throughout the Study, relative to its ability to compensate for wide ranges in available input data, may be somewhat misleading, especially if the log normalized data are "processed" through a series of "and" or "or" gates. It would appear that, in such cases, the Monte Carlo process would be expected to yield a point estimate similar to that attained through a straight additive or multiplicative process of input value median value failure rates, with an associated error factor. Although the Monte Carlo process is statistically correct, a further explanation of this process indicating the differences between it and the point estimate approach should be presented with regard to the evaluation of associated error factors. However, it should be noted that statistical techniques such as this, although appropriate analytical methodology, can never conclusively show that all critical pathways to an accident occurrence have been considered.

Chapter 6 of the main document presents a comparison of the nuclear accident risks to other societal risks. Although the Study does not make an absolute judgment on nuclear power plant accident risk acceptability, the comparative risk approach certainly implies an acceptability judgment to the average reader. EPA recognizes that the comparative risk approach is a first step in addressing this question; however, studies in progress by the EPA and others indicate that judgments on "risk acceptability" are extremely complex, with comparative risk evaluations representing only one of numerous inputs which must be considered.

Specific Comments

The question of applicability of the Study results to all current commercial water reactors is very pertinent. The discussion on page 27 appears to be the only place in the entire report that the question is considered, and only a brief general assessment is attempted. It is generally recognized that there are certain design differences in Babcock & Wilcox and Combustion Engineering plants as well as the Westinghouse plants of four-loop design (more common than the three-loop system selected). Similarly, the BWR containment design, in particular, has undergone two major changes (the Mark II and Mark III containments) since the reference design Peach Bottom plant, which would be expected to at least change the details of the containment response analyses. It would appear that the Study could benefit significantly by recognizing these design differences and presenting the necessary arguments which support the thesis that these design and response differences at the system design level do not have a major effect on the overall risk assessment. Further discussion appears warranted and any continuing analyses by the AEC to further verify this conclusion should be included.

On page 45, the safety improvement analogy with the aircraft and automobile industries regarding increasing safety with development is questionable. In both of these cases, safety improvements were accomplished by utilizing accident experience data. Although there has been significant variance in safety between particular designs in these industries, hopefully, similar significant differences will not be the case in nuclear plant safety designs. Furthermore, these industries have received increasing government control with the rise in concern over inadequacies.

On page 104, the argument for overprediction of fission product release from molten fuel appears to be partially contradicted by the discussion in the second paragraph under Meltdown Release Component on page 8 of Appendix VII. Similarly, the large surface area to volume ratio of the molten fuel described on page 119 should enhance the release of isotopes rather than "limit" it, as indicated. The discussion on page 109 of reduced doses associated with wind direction change conditions may be offset by increased evacuation difficulties. It is not clear if this has been considered.

CALCULATION OF REACTOR ACCIDENT CONSEQUENCES

Appendix VI

General Comments

Appendix VI appears to need substantial modification and information additions, especially with regard to the health effect calculations. The approach and methodology, although possibly adequate for the purposes of the Reactor Safety Study, should not be presented as the approach and methodology which calculates the consequences of accidents "as realistically as is now possible," as indicated in the USAEC's Interim General Statement of Policy on WASH-1400, dated August 28, 1974.

Although some of the factors affecting consequences are adequately discussed for the purposes of this report, there is no description of the overall calculational process which would permit a clear understanding of the relationships between the radioactive material releases, the dispersion, the population distributions, and the resulting health effects. Obviously, many refinements to the various calculation models are available. Those which were assessed and found negligible in effect for the purposes of this report should be discussed to give a better appreciation of the range of applicability of the calculation model used.

Specific Comments

The reasons given in Section 6.2 of Appendix VI for selection of critical radioisotopes do not support the omission of plutonium-241. Data presented by the USAEC in the draft WASH-1327, "Generic Environmental Statement on the Use of Recycle Plutonium in Mixed Oxide Fuel in LWR," on pages I-13, I (A)-2, and II-20, indicate that for exposure by inhalation of plutonium within a few years after its production in the LWR uranium fuel cycle, Pu-241 contributes more to the dose than Pu-239. Similarly, the data referenced by the Study (ICRP-II) and the plutonium isotopic mix of WASH-1327 indicate that Pu-241 contributes more to the dose than Pu-239 in the majority of the organ doses considered in Table VI-16 of Appendix VI, the proportion of their contributions depending upon the solubility of the plutonium aerosol.

The discussion of meteorological models and assumptions should be expanded to discuss the expected calculational differences incurred with the Study's use of a simplified model as opposed to the more conventional but complex models in general use. For example, it is inferred from the discussion on page 16 of Appendix VI that much of the meteorological frequency information is taken from greater heights than the release heights predicted in this Study. Since wind velocity generally increases with altitude, using such information will tend to decrease the estimated downwind dose levels. The acute health effects, therefore, could be underestimated. Furthermore, the uniform distribution in the crosswind direction used in the atmospheric model, as described in Section 6.4 of Appendix VI, is also likely to produce an underestimate of acute health effects, since the sector averaged dose estimates should be lower than actual peak doses. Finally, without consideration for wind meander, the constant angular widths chosen appear to broaden the plume more than would be expected (Ref. Figure A.2, page 408, of "Meteorology and Atomic Energy - 1968"), again contributing to lower peak doses and thus fewer acute health effects. It is judged unlikely, however, that any underestimate of acute health effects resulting from the treatment of meteorological information and the dispersion model is greater than a factor of two.

The model selected to account for the effect of evacuation on the calculation of medical consequences is described on page 31 of this appendix. An EPA report, "Evacuation Risks - An Evaluation," EPA-520/6-74-002, is referenced in support of certain assumptions used in the evacuation model. Although a number of parametric calculations relating to the evacuation model assumptions are presented in Table VI-21 (including a no evacuation case study), we believe the base case evacuation model to be overly optimistic.

The EPA report on evacuation risks was primarily directed at an assessment of the risk of death and injury, and the costs associated with past evacuations of population groups. The data and information utilized in this study were obtained by contacting persons and organizations involved with previous evacuations precipitated by natural or man-made causes. Factors which were hypothesized to influence the time required for the historic data base evacuations included: (1) time lapse before onset of incident, (2) availability of evacuation plans, (3) time of day, (4) weather conditions, (5) population size, (6) area size, (7) characteristics of the area, (8) conditions of roads, (9) nature of incident, (10) warning time, and (11) population density. No correlation with evacuation times could be determined for parameters (1) through (9). Similarly, since warning time was generally not separable from the time associated with accomplishment of the evacuation, no quantitative evaluation of this parameter was made. A correlation of evacuation times with population density, however, was performed assuming independence from other parameters. A trend showing an increase in time required for evacuation was indicated as population density decreased. In applying this conclusion to evacuations which may result from potential nuclear accidents, an element of caution needs to be exercised. It should be remembered that the data on historic evacuations generally include situations applying to small areas or in the case of larger areas, when there is a lengthy forewarning time. More significant is the fact that evacuation travel distances were almost always short and safe destination points were generally obvious.

Since the evacuations called for in the larger consequence accidents appear to involve evacuation areas of a few hundred square miles, the application of evacuation time requirements from the EPA evacuation study to areas of this size is questionable.

Although the Appendix VI discussion states that an outermost limit for evacuation of 20 miles was assumed, it is not clear how this evacuated population segment is treated in terms of actual dose received. Similarly, in the assessment of property damage on page 63, the effectiveness of the 10 rem calculated yearly dose as a basis for temporary evacuation is unclear since the expected dose rate as a function of time is not indicated.

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The assumption stated on page 63 regarding a first year projected dose of 10 rem as the criteria for determining the decision to evacuate may be unwarranted. A suggested value of 10 to 20 rem is cited, but the reference, although relevant, does not contain such a suggestion. In it, the recommendation is made that for small population groups, the use of evacuation as a protective action be considered if the anticipated exposure during 30 days might exceed a whole body dose of 2 rad or a thyroid dose of 10 rad. The reference suggests that under less favorable circumstances evacuation might not be considered as a protective action unless larger exposures were anticipated. The reference does not support the implication that 5 rem per year or less is acceptable because it is below the occupational dose limit, nor does it suggest 10 rem or any other projected dose as a criterion for decontamination.

The evaluation of health effects appears to require significant modification and information additions. Our preliminary indications are that if the BEIR Report (The Effects on Populations of Exposure to Low Levels of Ionizing Radiation, NAS/NRC, November, 1972) recommendations are followed, the assessed latent cancers are low by a factor of four (excess mortality by a factor of two), genetic effects are low by a factor of five, and acute deaths and acute illnesses are low by an undetermined factor. Although such changes in calculated health effects do not necessarily alter the results of the Study, either these changes should be made or the rationale for utilizing other assumptions should be justified. The following discussion highlights our evaluation in this regard.

Section 6.6 appears to totally neglect the 2% of the population in utero. A list of applicable references related to this subject is included in the additional comments section of our review. This rather important segment of the population should be considered with regard to the effects of radiation.

On page 34 and 48, a statement is made that the BEIR risk estimates are upper limit values. In fact, the BEIR Committee estimates, as they state, may be too high or too low. Although the value of zero could not be excluded by the data, use of this number was rejected on a number of cogent bases (pages 2 and 88 of the BEIR Report). Also, on pages 34 and 48, the somatic excess death risk used (50-165 deaths/10^6 man-rem) only reflects the range of absolute risk estimates from the BEIR Report. Consideration should be given to the estimates which, in combination with the absolute risk estimates lead to the BEIR Committee "most likely" numbers (150-200 deaths/10^6 man-rem). Using BEIR "most likely" numbers for a lifetime plateau instead of a 30 year plateau for expression of effects, the EPA estimate of 200 excess deaths/10^6 man-rem has been developed (page 167-174 BEIR).

The BEIR Committee genetic effects estimates referred to on page 35 may be construed as meaning that 10^6 man-rem would produce 10-100 dominant diseases and 1-100 congenital anomalies in the first generation after exposure. However, this is 1/5 of the total impact expected. An additional 40-400 dominant diseases and 4-400 congenital anomalies should be attributed to future generations (page 53 BEIR).

The reference for the numbers in rads used as criteria for estimating acute effects is not presented. The average dose which will cause fatalities to 50% of the people so exposed in 60 days is given by Lushbaugh, Comas, Edwards, and Andrews (Sect 17 in AEC CONF 680410, 1964) as about 285 rads. These authors estimates for the dose which will cause fatalities to 10% of the people so exposed, LD_{10}, is of the order of 75-80 rads with a range of about 40-120 rads. This estimate is of the same order of magnitude as the estimate of less than 5% mortality in the dose range of 40-140 rads given in NCRP Report #29. Therefore, the statement on page 37 that there is little chance of death from doses below 100 rads appears somewhat optimistic for estimating the possible effects on a large population. A more accurate estimation of effects would be made using an appropriately justified probit analysis with perhaps a cutoff at 40-50 rads.

On page 50, a statement regarding the deleterious genetic changes expected per 10^6 man-rem of exposure is presented, which also appears to be a misinterpretation of the BEIR Report. The BEIR Committee estimated that the average mutant persisted in the population for five generations not "...in the first and also in all generations..." Therefore, the total increments shown in Table VI-14 should be five times greater.

Similarly, the quotation from the BEIR Report (page 91) appearing on pages 52-53 is truncated to an extent that, in our opinion, a misinterpretation of the BEIR Report results. The paragraph quoted continues: "By extrapolation, it can be estimated that the number of deaths per 0.17 rem per year in the entire U.S. population may range roughly from 3,000 to 15,000 with the most likely value falling in the range of 5000 to 7000 (or 3500 per 0.1 rem per year)." Utilizing this last estimate, the excess mortality from all forms of cancer calculated in WASH 1400 would be almost doubled.

Possible clinical effects from acute radiation exposure other than death and radiation sickness are discussed only briefly on page 46. For example, temporary aspermia in the male has been observed following exposures as low as 12.5 R. The personal trauma of being unable to reproduce or of it being recommended that no attempt be made to conceive a child for some extended period after exposure is not negligible, at least for normal "peacetime" operations. Furthermore, the disruption of the homeostasis of the finely tuned endocrine system, while possibly amenable to hormone replacement therapy, does not necessarily represent insignificant individual trauma or financial burden. Therefore, a significant expansion of this presented discussion appears warranted.

ACCIDENT SEQUENCES, REACTOR MELTDOWN PROCESSES AND RADIOACTIVITY RELEASES Appendices V, VII and VIII

General Comments

These appendices, which follow various accident sequences through the meltdown process and associated releases of radioactivity, represent a significant effort to quantify the consequences of reactor meltdown accidents. It is recognized that to present a meaningful discussion of the many accident sequences evaluated, to relate these sequences to the timing and physical processes associated with a reactor meltdown, and to predict the resulting radioactivity releases via several containment failure mechanisms is a formidable task both technically and documentarily.

Of these three appendices, Appendix V (possibly because it pulls together much of the information presented in Appendices VII and VIII) appears to require some additional effort to resolve apparent inconsistencies and to supply additional information on accident sequences other than the large LOCA. Furthermore, the Study should highlight and expand the sensitivity analyses on ECCS functionability and the evaluation and significance of the various containment failure mode probabilities.

Our comments on both Appendices VII and VIII are dealt with in the specific comment section which follows.

Specific Comments

Appendix V

One problem in reviewing Appendix V involves apparent inconsistencies between the various tables which relate accident sequences to release categories. For example, on pages 21 and 24 (Tables V-3 and V-4), it is not clear how these lists were compiled. Both tables do not include some of the dominant large LOCA sequences from Table V-6 (e.g. AF-α and ACD-α in category 1, and AF-δ in category 3) but do include sequences which are not considered dominant (e.g., ACDGI- α). Based on the discussion (page 140), it is also not obvious why sequence ACDGI -α is classified as release category 1 instead of category 3. Furthermore, in comparing the probabilities given in Table V-6 with the relative containment failure mode probabilities listed in table 2 page 124 of the attachment, certain sequences listed as "other large LOCA accident sequences" appear to be significant contributors to a release category probability (e.g., category 2, AHF - α, $3x10^{-10}$; AB-δ , 4×10^{-11}). If these contributions are numerically correct, the sum appearing at the bottom of the table must only represent the sum of the listed dominant sequence probabilities. Since the large LOCA's do not dominate the probabilities of Table V-16, information similar to that presented in Table 2 of the attachment, page 124, applicable to small LOCAs and transients would appear pertinent for inclusion in this appendix.

We would like to emphasize at this point that inclusion in a release category probability estimate of a 10% contribution from adjacent release categories adds considerable conservatism to certain summed release category probabilities; however, an attempt should be made to correct and clarify the interpretation of these summary tables.

In the discussion of the smoothing of release category probabilities, p. 50, it is not clear how the smoothing technique necessarily swamps any common mode failure contribution. The presentation would also be clarified if even just an illustration were included which would show the bar chart in Figure V-1 reversed in relation to the severity categories.

Considering the interest attached to the ECCS functionability, the discussion on pages 52-55 is especially pertinent. This sensitivity analysis discussion might be considerably improved by not only relating the ECF contributions to overall release category probabilities but also to the "large LOCA" contribution. This latter relationship would show a larger percentage contribution. For example, given a large LOCA (A) Followed by ECF (E), one accident sequence would be AE- ε with the same consequences as AD-ε (category 7). The probability would be AE- ε = $(1 \times 10^{-4})\ (10^{-2})\ (\approx 1) = 10^{-6}$ assuming the high end of the ECF failure rate. Although this and other sequences would have a moderate influence on large LOCA release categories, the limited impact on the overall release category probability would be highlighted. Since the ECF failure probabilities are of general interest, it would appear appropriate to identify the rationale for assuming the failure occurrence range utilized (10^{-2}- 10^{-5}). Considerable confusion is also caused by not including ECF sequences in Table V-16 while including such sequences in Table V-6.

With regard to the BWR transient tree quantification on page 68, it is not clear from this discussion, in conjunction with Table V-19, which transients were slow enough such that credit for reserve shutdown can be taken .

In attachment 1, Table 2, certain sequences are shown with "containment rupture - vessel steam explosion" failure mode probabilities of zero which are nevertheless estimated as 0.01 in Table V-6. Since similar tables are not included for S_1 and S_2 initiating events, the relationship between the various containment failure mode probabilities shown in Table V-7 and V-8 cannot be determined (e.g., the relationship between S_2 C-δ and S_2 C-α).

Appendix VII

The information contained in Appendix VII is well presented, sufficiently documented, and based on our preliminary review, presents a reasonable appraisal of the extent of radioactivity releases.

In discussing the meltdown release component for alkaline earths and noble metals (p.p. 11; 13), the probable values selected appear somewhat low if consistency with the selection basis of other released components (e.g., halogen, alkali metals) is to be maintained. In fact, the text, in discussing the alkaline earths, indicates a release range of 2-20% and suggests that the probable value should lie in the upper portion of this range, yet selects 10% as a most probable value.

On page C-1, last sentence, it is not clear what is referred to by "...the LOCA's postulated; i.e., successful ECC and recovery," since the Study is concerned with many LOCAs in which successful ECC and recovery are not assumed.

In outlining the accident sequence and core response on page C-2, the basis for the 100% rod failure at a maximum clad temperature of 2200 ° F should be stated since this failure value appears to be quite conservative in view of vendor calculations (eg. Surry, Final Safety Analysis Report). Also, in the discussion of six critical points involved with the evaluation of the LOCA prompt release fission product source-term, the term "release coefficient" (escape fraction) should be made consistent with nomenclature used elsewhere in the report.

In Appendix K, p. K-19, it is not clear if the text is implying that a potential important pathway for release of fission products, between the containment shell and cofferdam, was not considered in the Study.

Appendix VIII

The discussion under "Limitations" on page 3 of Appendix VIII contains a disclaimer regarding the potential non-applicability of "these studies" (presumably core meltdown studies) to other PWRs and BWRs. As mentioned previously under the specific comments on the main volume of the Study, the discussion on this topic should be expanded.

In discussing the basic assumptions for the analysis of degraded accident behavior (p. 7), the basis for assuming that "core melting would take place without significant metal-water reaction and that there would be no possibility of steam explosions in the reactor vessel" under conditions of accumulator and pumped ECI failure needs further explanation to account for the possibility of residual water being left in the vessel from the blowdown process. With regard to the accident time scale, the 10- 11 second time quoted for essentially complete primary system depressurization and the time of accumulator discharge appear a factor of 2-3 too short compared to the results in the paper, "Comparison of Thermal-Hydraulic Response of LOFT and a Large PWR to LOCA Conditions," authored by P. Davis and J. Ductone, presented at the topical meeting on water reactor safety. CONF-730304, March 1973.

On page 12, the dismissal of the potential for a large energy release from a steam explosion between the molten core and water laden gravel seems to be contradicted by the Armco Incident described on page B-2.

In describing containment response (p-13), the assumption of LPRS cavitation at the time of containment failure seems pessimistic, if Regulatory Guide 1 is followed. The guide states, "Emergency core cooling and containment heat removal systems should be designed so that adequate net positive suction head (NPSH) is provided to system pumps assuming maximum expected temperatures of pumped fluids and no increase in containment pressure from that present prior to postulated LOCA."

The meltdown sequence discussion, which includes CHRS and LPRS failures (p. 17), describes the molten core vessel penetration and interaction with the water in the reactor cavity and the CSRS water. Our understanding is that the CSRS water should not be expected in the reactor cavity except for possibly a small amount of leakage. If this interpretation is correct, all sources for reactor cavity water need further clarification. Similarly, on page 21 it appears that CSIS is assumed not to deliver water to the reactor cavity while the opposite is true for CSRS.

The assessment of containment failure mode probabilities includes the probability of containment failure resulting from a steam explosion estimated as P = 10^{-2} (+1,-2). Since we are not aware of any discussion which indicates that this probability is sequence dependent, the probabilities associated with certain sequences in Table V-16 are not understood (e.g., $S_1D-\alpha$, 3 x 10^{-8} while $S_1D-\alpha$, 6 x 10^{-6} and no S_2 $C-\alpha$ which should be at least 2 x 10^{-8} based on S_2 $C-\alpha$ of 2 x 10^{-6}). The latter example may be eliminated because containment overpressure failure occurs before initiation of core meltdown. If this or other sequences are logical exceptions to the containment failure probability associated with vessel steam explosion, the exceptions should be discussed.

The assessments of containment failure probabilities from hydrogen combustion or overpressurization both are strongly dependent on the assumed normal distribution of containment failure pressures of about 100 psia with a 15 psi standard deviation and containment melt-through time (for which the meaning of the skewed distribution, 18(+10,-5) hours, is not clear). Given the information in figures 4 through 9, general correlation with containment failure mode probabilities listed in table 2, attachment 1, Appendix V could be observed. However, such was not the case for sequences AHF-δ and AB-δ . It would be helpful if the text provided an example of such a calculation, which would define the various probabilities listed in the text.

Additional clarifying remarks would seem appropriate at several points within Appendix A, which discusses the thermal evaluation model. On page A-16, the assumptions used for the core temperature distribution and vessel water inventory following blowdown should be stated and justified.

Regarding the fission product release fraction equation, the basis referred to should be specifically referenced. Similarly, the reasoning for assuming no change in steam properties due to hydrogen mixing should be presented. Under the heading "Convective Heat Transfer" the values chosen for T_W and h_B should be discussed since it would appear that h_B should vary with Q_{DGR}, ΔT, and pressure.

On page A-33, a question arises as to whether vessel failure can occur by fracture due to thermal stress occurring when the molten core contacts the lower vessel head.

It is not clear on Page A-36 if the continued addition of water on top of the core melt could cause a steam explosion similar to the East German incident described on page B-3.

The containment failure mode evaluation presented in Appendix E considers several factors which could affect the ultimate containment strength. Further discussion or clarification of the potential significance of these factors on the assumed 100 ± 15 psia failure pressure appears warranted. Since the assumed failure pressure could alter the containment failure mode probabilities for several accident sequences, an indication of the sensitivity of the release category probabilities to a change in the assumed containment failure pressure should be provided.

DEFINITION OF FAILURE DATA AND PATHWAYS

(Appendices I, II, III, AND IV)

General Comments

Our review of Appendices I and II has, for the most part, been limited to questions regarding the treatment of (1) specific failure pathways which are not acknowledged in these appendices or (2) the rationale for dismissal of other failure pathways or their relationship to assigned containment failure mode probabilities. Review of certain sections of Appendix II is presently anticipated in our continuing review of the Study.

In our limited review of Appendix IV, the role of the "common mode failure" in the overall risk assessment process is difficult to assess. Some methods, quantification techniques, causes and results are discussed in Appendix IV, but the material necessary to properly understand the total role and significance of common mode failures and to determine that a reasonable degree of completeness has been developed appears to be spread through Appendices I, II, III and V. Many summary statements in the earlier and later sections of the report assert the significance of common mode failures without quantification or reference when, in fact, the needed material is in other appendices. Further cross-referencing to such analyses and quantifications which support the assertions of this appendix could resolve these concerns.

Specific Comments

Appendix I

In the LOCA functional event tree development (p. 13 footnote), it appears that the containment building purge system has a probability of failure which is not acknowledged. Similarly, the possibility of containment overpressure failure prior to core melt which is treated in subsequent appendices, is not included in the discussion on page 31.

In the development of the PWR small LOCA event tree, S_2, p-132, it would seem possible that vessel melt-through could occur while the primary system pressure is above the accumulator injection pressure. After vessel melt-through, the primary system pressure would be rapidly reduced, allowing possible accumulator water injection onto the molten core, potentially causing containment rupture from a steam explosion. It could not be determined if the containment failure mode probabilities for S_2 LOCA's considered this possibility.

With regard to PWR reactor vessel rupture, p. 141, it is not clear how the polar crane presents an effective missile barrier for the entire upper portion of the containment.

On page 145, the reason for not considering rupture of steam generator tubes and subsequent overpressurization of the secondary system with potential for rupture outside the containment, should be stated. Similarly, on page 159 it is not obvious why the situation of automatic trip failure occurring with loss of electric power sequence was eliminated from consideration.

Appendix II, Vol. 2

The discussion of the electrical power system, offsite common mode failures (p. 33), does not specifically indicate whether earthquakes have been considered in assessing common mode failures, especially for power subsystems which are not specifically designed for earthquake response (diesel fuel system). Similarly, in the text on page 35, a discussion of how the failure analysis of the diesel generator system accounts for the failure modes discussed would be an important addition to this section.

The evaluation of the reactor protection system (RPS) dismisses the impact of a pressurizer vapor space rupture on the RPS (signal initiation) failure probability. Since it is possible for the low pressurizer level signal not to function due to frothing, the effect of such a failure should be addressed. On page 155, under CSIS failure modes, it appears that the Refueling Water Storage Tank (RWST), "suction line plugged," should also be listed under single failure resulting in unavailability of RWST water.

The Consequence Limiting Control System (CLCS) description on page 174 is confusing since it appears from this discussion that the operator may not be able to switch from CSIS to CSRS until the containment pressure falls to -0.5 psig (such a pressure may not occur in time to permit successful switch of these systems).

The results of analysis of system interfaces under Low Pressure Injection System (LPIS) indicates, on page 280, that momentary unavailability of water at the start of LPIS pump operation is not considered a failure, while unavailability of water to the CSRS is considered a failure for the same reason (Appendix I, p. 102). A clarification of this situation appears warranted. Also, in listing the single failure-failure modes for LPIS (p. 284), consideration should be given to pipe ruptures between J3 and either V4 and V5 (figure II-53) and to RWST pump suction drain plug.

In the examination of potential faults for the Low Pressure Recirculation System (LPRS) on page 498, it is not clear that pipe ruptures between P1, P2 and J8 will not cause system failure. The flow would split between the path to the cold leg and the rupture and, depending on relative flow resistances, the delivery of water to the cold leg may not achieve the necessary 300 gpm.

Appendix II - Volume 3

In the evaluation of the BWR electric power system, an assumption is made on page 27 that all emergency buses are available immediately prior to a LOCA based on the Technical Specification requirement that the reactor be shutdown if an emergency bus is not available. However, it appears there should be a finite probability that all emergency buses are not available which should be dependent on the failure probability of the failure detection system. Also in discussing offsite power common mode failures, the omission of earthquake as an initiating event should be addressed.

Discussion on page 134 relates failure of vacuum breaker valves in the open position to the defeat of the vapor suppression function. It would seem appropriate that the assumptions regarding the two or more valve failures required should be justified with calculations or referenced to pertinent information. Similarly, the assumption on page 243 that rupture of branch piping of 2-inch diameter or less will not significantly affect core spray injection system operation at the time of a LOCA or during injection requires justification.

Appendix III

This appendix on failure data is well written, well organized and appears to be appropriately integrated into the Study. In our continuing review, reflected in the attached work plan, EPA does intend to perform a selective review of the failure rate data base.

Appendix IV

Our review of Appendix IV, to date, has been limited, especially with respect to the analysis and quantifications applied in the Study; therefore, our comments at this time are very general in nature. The concept and influence of the "common mode failure" appear to need additional development. A distinction should be provided between the causative effects of certain common mode failures in initiating accident situations vs the influence effect of common mode failures resulting during an established accident sequence (e.g., the influence of a check valve slam and subsequent water hammer damage as a common mode initiator vs the consequence of such an event occurring during an accident in progress).

The completeness of the consideration given to potential sources of common mode failures also appears to require some expansion. Although, through searching in other appendices, it is evident that particular types and areas of common mode failures are considered, sources, such as the requirement for pump inlet subcooling for emergency coolant recirculating systems, pump bearing lubrication systems, instrument and component service water and air, heat tracing for plateout prevention, control room and cable tray fires, drain plugging of storage water systems, etc, ought to be discussed and treated in Appendix IV to support the claims and assertions developed.

Finally, the treatment of the method of screening for relevant common mode sources discussed on pages 18 through 39 appears credible but is unsupported and in need of tabular listing of (or reference to) the "numerous" types of sources considered in order to demonstrate that the method is indeed comprehensive in identifying all conceivable component, system, and operation vulnerabilities to common mode failures. The examples cited are useful but create questions of "what else," "how many," and what does a complete list look like.

DESIGN ADEQUACY

(Appendix X)

General Comments

This appendix does not appear to be tied in with the rest of the Study. There is no mention of how the results of this Study were utilized in the risk assessments. Since Appendix X indicates that a significant number of the systems examined were either not properly qualified, not properly analyzed, or didn't meet current standards, it would seem very important that these deficiencies be readily traceable to the quantitative risks, or that they be shown to be peculiar to the plant analyzed.

Specific Comments

The section on seismic loads (p. 47) appears incomplete in that current design response spectra were not evaluated for the structures and equipment. On pages 52 and 57, it is stated that the current spectra would increase seismic loads (by as much as a factor of 2). It is not clear what these increases mean relative to the general seismic vulnerability of the 100 plants and what risks are associated with the increases.

SUMMARY REPORT

General Comments

The summary document is a relatively well written volume, which satisfies its intent through a question and answer format. Our comments on certain quantifications of assessed impact are incorporated into our review of the Appendix VI volume. Of particular interest was the discussion comparing the Study predicted consequences with the earlier WASH-740 evaluation. It would appear that the significance of the four factors leading to differences in the two studies is substantial. Plume rise and evacuation and possibly population, as treated in WASH-1400, have relatively little impact on consequence when compared to the effect of the differences in assumed release of radioactivity to the environment. The Study indicates that given a PWR core meltdown event, a chance of only about one in one hundred exists that the resultant containment failure mode will be other than melt-through with its relatively insignificant radioactivity release. A somewhat similar case exists for the BWR meltdown event where a chance of only one in ten exists that the containment failure mode will be other than containment isolation failure in the drywell with, again, a relatively insignificant release of radioactivity. It would appear appropriate that the discussion of this variable in the summary document should be expanded.

Additional Comments

Main Volume

Clarifications

1. Page 122, Table 5.2 - It is not clear why AB, ACHF, S_1B and S_2B do not lead to containment failure by overpressure since loss of containment heat removal should lead to overpressure failure.

2. Page 126, Section 5.3.2.1 - The definition of a large LOCA being a rupture equivalent to a hole greater than 6 inches in diameter is not consistent with the definition used by vendors, AEC-Regulatory, etc., which is a 0.5 ft^2 (9" hole). It is not clear why a different definition was chosen here.

3. Page 154, item (1) - The SL-1 accident was a military power reactor nuclear accident which resulted in 3 fatalities. It appears that the statement ignores the SL-1 accident.

4. Page 216, last sentence in Section 6.4.7 and Figure 6-10 - The statement that the calculated probability of a dam failure resulting in 10,000 fatalities "...agrees with the extrapolation of the data..." does not appear justified. A straight line can be drawn through the three data points (as was done in Figure 6.9), and, if anything, an upward inflection of the curve is indicated by the data, rather than downward as drawn, to include the calculated point.

Editorial

1. Page 146, Section 5.4.4 - The source for the probability of aircraft impact accidents should be referenced.

2. Page 150, Section 5.4.6 - Near site explosions, which must be considered for reactor sites, are not mentioned.

3. Page 200, Section 6.4.1, 1st sentence - The reference does not agree with the reference at the end of Table 6.8.

4. Page 204, reference 1 - This reference appears incorrect, "...North Atlantic Hurricanes..." since the Galveston hurricane is apparently included (#1, described on page 200).

5. Page 205 - The average number of tornado fatalities is stated as 118 while the division indicated yields a value of 46.

Appendix I

Clarifications

1. Page 83, Figure I-13 - It is not clear why the success path for containment leakage is chosen as the drywell and the failure path, the wet well. Wet well leakage should produce the lesser consequences due to fission product scrubbing in the torus (see Appendix I, page 37).

2. Page 102, 3rd sentence, and page 134, item F. - The basis for the CSRS failure assumption is not clear since CSRS should eventually operate.

3. Page 199 - Further justifications of the unanticipated transient probability of 10^{-5} per year should be presented.

4. Page 205, Figure I-28 (also Footnote 1, page 207) - The RPS failure probability for unanticipated transients (Part C) has been increased from 4×10^{-7} to 4×10^{-6} to account for the fact that only the scram system may be effective for reactor shutdown. This reduction does not agree with the fault tree at the bottom of page 68, Appendix V, which assigns the failure of RPS to scram a value of 1.3×10^{-5}.

Editorial

1. Page 45, 3rd paragraph, 1st sentence - It appears that CR-VSE should be CR-CSE.

2. Page 77, Figure I-10 - The LPIS is missing from the ECI segment of this figure.

3. Page 198, 1st paragraph - The apparent distinction between a transient which causes a LOCA and a transient which causes a ruptured reactor coolant system is not clear.

4. Page 222, 4th sentence under RHRS - This sentence is ~~not~~ complete,

5. Page 233 - These footnotes appear to be used in Table I-13, but the heading does not match the heading for Table I-13.

6. Page 259, item 5 - This item appears out of place in that it is not a "...design feature provided to keep the likelihood of loss of pool water small..."

Appendix II, Vol. 1

Editorial

1. Page 11, last paragraph - This disclaimer paragraph seems to indicate that if data did not exist for a particular system failure contribution, it was not considered.

Appendix II, Vol 2

Clarifications

1. Page 253, item 3 at top of page - There does not appear to be any basis for the assumption that pipe ruptures of 2 inch diameter or less will not cause failure of accumulator injection.

2. Page 385, Introduction - It is not clear if the SICS analysis also applies to the small break case.

3. Page 490, top of page, and page 529, 1st paragraph - It is not clear how realignment of the LPR system to the hot legs will prevent an "undesirably high boron concentration or accumulation of residue and debris in the core that could result from continuous boiling." LPR system water injected in the hot legs will enter the upper plenum, run down the outer (cold) core and core structure region into the lower plenum, and be available for boiling in the hot central core region.

4. Page 490, top of page - It appears that closure of V_{10} is also required to effect the realignment.

5. Page 501, 2nd paragraph - It is not clear why air suction from the RWST occurs for this failure in view of the discussion under item (2), page 498.

Appendix II, Vol. 3

Clarifications

1. Page 92, Section 3 - It is not clear why Q_{MED} (1.3×10^{-5}) is considerably different from the RPS unavailability shown in Figure II-131 (2.47×10^{-6}).

Editorial

1. Page 359, item 2 - The use of the term "suppression chamber" is inconsistent with "wetwell" and "vapor suppression system" used elsewhere in the report.

Appendix III

Editorial

1. Page 187, first line - The bibliography section mentioned here appears to be missing.

Appendix IV

Clarifications

1. Page 43 - Results of the susceptibility analysis are presented but no specific reference is given to where the analysis is presented and the specific fault and event trees to which it was applied.

Editorial

1. Glossaries and definitions are sorely needed for this appendix, not only to track the latter sections in relation to Appendix II, but to understand the distinctions between the PWR and BWR treatments.

2. Page 8 - The treatment of ideas at this early stage in the appendix requires the reference to other unspecified appendices in order to understand the terms used and messages developed. An introductory tutorial treatment with a description of the other appendices which intimately interface with Appendix IV is needed.

3. Page 8-15 - This section could benefit by specific cross references, examples and limited numerical results to give significance and meaning to this important portion of the report.

4. Pages 40-41 - The list of "classes of potential common mode mechanisms" could benefit by a sub-category of items under each major topic to provide an index of completeness, e.g., where would failure causes fall for wearout due to exercising a given component, or for partial or delayed performance due to degradation from lack of service, or for transient behavior of a component (check valve water hammer).

5. Pages 40-63 - Although Sections 3.3 through 4.0 portray a reasonable description of the methods applied to the "quantifications" in the study, the interpretation could be considerably aided by examples with numerical results or tabulations, such as that of Table IV-4 on coupling probability.

6. Pages 65 and 87 - These two sections are intended to treat PWRs and BWRs separately and this should be stated in the introductory paragraphs.

7. Pages 65-98 - This Section, "Summary of Results," acknowledges the performance of the "fault analysis" in Appendix II and from those results identifies selected "sequences in the event tree...chosen because...(of) some potential susceptibility for common modes" and develops "impact" conclusions as "insignificant," "minor impact," etc. The support for and meaning of these conclusions should be identified.

The event sequences selected for the follow-on discussions appear without comparative discussion to other cases which have been dismissed. Although these discussions improve one's insight to the "controlling" common mode sequences, tabulations or some form of overall results presentation should be developed to enable the reader to gain a "feel" for the relative influence or "impact" of other sequences which could be important to plants of newer design than those chosen for analysis. The companion treatment given to the BWRs (page 87), although different in style, is equally obscure in portraying understanding and confidence that the treatment of common mode failures is comprehensive and complete.

APPENDIX VI

Clarifications

1. The description of the release and dispersion calculation in Appendix VI appears sketchy in that there is not a clear description of the radioactive material release magnitudes as a function of time over the release durations presented. Thus, any interaction of the airborne release with the population being evacuated cannot be evaluated. The description suggests that the fraction of core inventory released is modeled as a uniform release over the indicated duration of release. An alternative model could be a distribution of discrete releases as shown in Figure J-8 of Appendix VII. A clarification of this subject is in order.

2. The discussion of the consequence calculation and population distribution patterns of Appendix VI does not describe the model of the population distributions used for calculation of consequences within 70 miles; i.e., it is not discernible from the information presented whether the sector population, originally obtained as a function of distance from the reactor, was averaged over the first 70 miles or averaged over segments of sectors using differences in the cumulative populations from Table VI-6, or whether some other distribution model was used.

3. A more careful explanation of the population averaging method on page 24 would be helpful. In particular, the top 1% sectors are reflected in the peak case consequence results. The range of populations averaged into the top 1% would clarify the nature of the top population category.

4. The application of the plume broadening for meander over extended periods of time, described in Section 6.43 of Appendix VI, needs to be specified more clearly. Table VI-2 shows categories PWR 6 and PWR 7 having a duration of release of 10 hours and all other categories having shorter releases; it is not clear whether categories PWR 6 and PWR 7 are the only categories having "releases that last for many hours," i.e. categories to which the broadening was applied, or whether the broadening was applied to shorter releases as well.

5. Because this appendix does not present the necessary information regarding individual organ or whole body doses as a function of release category and downwind distance, several questions arise as to the significance of certain omissions from Table VI-16; namely, (1) lung dose contribution from noble gas inhalation, (2) consideration of Pu-241, Am, Cm, and U releases, (3) releases of longer lived isotopes, such as I-129 and H-3, and (4) any possible significant release of activation products.

6. With regard to the evacuation model, clarification is needed of the manner in which the warning time for evacuation T_j (time between awareness of impending core melt and leakage for accident type j) was determined. It is observed that, in Table VI-2, this time is constant for each reactor type and independent of the containment failure mode, and also that for release category PWR I, awareness of impending core melt is immediate at the outset of the accident.

7. Page 36, Section 6.6.3. - This section is based on available data and is apparently extended for standard man only. The uncertainties in the estimates, particularly as they apply to differences in age and state of health, should be at least underscored and, if possible, explored further.

8. Page 37 - The listing of peripheral blood element response should be compared to data given by Wald (Chapter 23, Haematological Parameters after Acute Radiation Injury, pp. 253-264 in Manual on Radiation Haematology, IAEA Technical Report Series No. 123, 1971).

9. Page 47, Section 6.6.4.4. - Reference and justify assumptions, particularly the "...slightly increased number of induced mutations." If a value judgment is to be made, a frame of reference must be established.

10. Although reference is made to the BEIR Report, the discussion regarding Table VI-13 is misleading. This table, taken from p. 171 of the BEIR Report, refers to the "...absolute risk for those aged 10 or more at the time of irradiation..." This is neither the complete estimate of the BEIR Committee nor the only population considered.

11. Page 49, Section 6.6.4.3 - This section does not mention the rather generalized "increased ill-health" considered in the BEIR Report.

12. The discussion of thyroid illness on pages 53 and 54 appears to need considerable clarification. In particular, the apparent treatment of production of nodules as an illness requiring a surgical process is not understood. For an estimate of nonfatal malignancies, reference to the BEIR Report would seem appropriate.

13. Page 54, first paragraph - The assumptions on incidence of nodule formation following thyroid exposure discussed on page 54 should be justified. For example, data in Reference 42 of the subject draft report suggest that, in a mixture of external and internal radiations, gamma and beta exposures are equivalent. The BEIR Committee points out studies evidently showing a species difference in response to beta irradiation of the thyroid and also points out the problems in some available human and I-131 data (a thyroid ablating dose is used).

While Reference 42 does mention thyroid nodularity incidences ranging from 0.47% to 1.6%, it should be pointed out that the 1.6% incidence was in a population of 30 to 59 years of age and 0.47% was in a general population. The 0.36% to 1.7% values in controls in various studies reflect small numbers in the populations and, perhaps, the regions of the country from which the populations were derived.

Lilien, et al (AM Lilienfeld, M. L. Levin and I. I. Kessler, Cancer in the United States, Harvard University Press, 1972), suggest a thyroid cancer incidence rate of $40/10^6$ persons based on state tumor registry data. Even if the ratio of fatal to occult cancers of 1 to 100 (ABCC Tech Report 25-68) is used and the incidence of $40/10^6$ thyroid cancers is considered fatal, the total incidence of thyroid cancer would be $4000/10^6$ persons. The relationship between these occult carcinomas and the total number of nodules has not been established yet, but some nodules are occult thyroid carcinomas. The nodules, as pointed out in Reference 42, represent malignant and benign tumors, but also nodular goiter, Hashimoto's thyroiditis, colloid diseases, local hyperplasia, local lymphnodes, etc.

14. Table VI-15 is somewhat misleading in that it apparently refers only to acute or subacute fatality and to "illness" in which thyroid should not be included since nodularity is not an "illness." The table does not include all effects, e.g. effects of pituitary injury or carcinogenesis, aspermia, etc.

15. References pertaining to in utero acute fatality and acute somatic injury are as follows: Evaluation for the Protection of the Public in Radiation Accidents; IAEA Safety Series # 21, IAEA Geneva (1967); Nokkentved, K. Effect of Diagnostic Radiation on the Human Fetus; Munksgaard, Copehagen (1968); Griem, M. L. The Effects of Radiation on the Fetus; Lying in: Journal of Reproductive Medicine 1:367-372 (1968); Hammer-Jacobsen, E. Therapeutic Abortion on Account of

X-ray Examination During Pregnancy; Danish Medical Bulletin. 6:113-122 (1959); Brent, R.L. and Gorson, R.O. Radiation Exposure in Pregnancy Current Problems in Radiology Vol. 215 (1972); Graham, S., Levin, M. L., Lilienfeld, A.M., Schuman, L.M, Gibson, R., Dowd, J.D., and Hempelmann, L. Preconception, Intrauterine, and Postnatal Irradiation as Related to Leukemia. pp. 347-371 in Epidemiological Approaches to the Study of Cancer and Other Chronic Diseases National Cancer Institute Monograph 19, NCI (1966).

16. There is also no indication that individual organ doses have been aggregated as "organ-rem" for summation in the estimate of "latent" cancers and genetic effects. Estimates of some isotopes and the distribution of organ doses and variations with age can be obtained from such publications as ICRP-17 (ICRP Publication #17, Protection of the Patient in Radionuclide Investigations, Pergamon Press. 1971).

17. Page 55, Section 6.7.3 - The use of ICRP-2 dose models, while defining what was done, does not seem adequate in light of advances in the field of physiology and dosimetry. As pointed out by Eve (I.S. Eve., "A Review of the Physiology of the Gastrointestinal Tract in Relation to Radiation Doses from Radioactive Materials," Health Physics 12:131-161, 1966) residence times and mass of contents for the GI tract used in ICRP-2 may be in error by factors of 2 or 3 in various segments and the values used for the stomach may be in error by a factor of 24 when residency time for inhaled material is being evaluated.

Dolphin and Eve (G.W. Dolphin and I.S. Eve, "Dosimetry of the Gastrointestinal Tract", Health Physics, 12:163-172, 1966) suggest that differences of the order of a factor of 2 result, when a more sophisticated GI tract model is used rather than the ICRP-2 model.

Eve also made pertinent comments on the dose to the ovary from GI tract contents and the insensitivity of mucosal cells to radiation exposure at a depth of less than 140 microns.

The lack of information on particulate aerosol characteristics of the expected releases used in this section precludes applying the more accurate Task Group Lung Model or determining the extent of departure from the simple ICRP-2 model which would be expected. However, the current biological half-times for the various isotopes could be employed.

18. In the evaluation of damage from an accident, the health effects and dollar costs appear to be considered as mutually exclusive. This fails to consider the dollar costs of health effects. There is of course, the obvious cost of lost productivity but it is also noted, for instance, that thyroid nodules are passed off as being surgically treatable with no consideration as to the dollar cost of that treatment.

19. In Section 6.8.4, Non Core Accidents, Table VI-23 appears to over estimate the consequences by up to three orders of magnitude.

Editorial

1. In Section 6.4.4, for the phrase in parentheses, "vertical velocity toward the ground," substitute "ratio of the ground concentration to the integral over time of the adjacent air concentrations." This substitution will avoid furthering the false impression that the deposition velocity is indeed the vertical velocity toward the ground.

2. The PWR 7 category description on page II of Appendix VI needs a few more words of clarification, since the sprays do not act on the leakage occurring upward around the containment.

3. In the second paragraph on page 14 of Appendix VI, insert the word "acute" before the word "illness."

4. On page 14, the sentence "It was found, in particular, that the wind blew 0.1% of the time toward the 0.1% highest population density sector" needs clarification. The explanation on page 110 of the main volume is much clearer.

5. In Section 6.5.1, the reference to the isolated Idaho Falls site is of questionable interest, since Idaho Falls is not the site of any commercial nuclear power plant.

6. Table VI-6, on page 28 of Appendix VI, needs correction in that it shows, for categories 11 and 12, that the cumulative population decreases as the distance increases from 2 miles to 5 miles.

7. Page 32 - Experience with human radiation effects is not small and includes much more than Japanese data. The experience with acute effects is much less.

8. Page 35, Section 6.6.2 - The question of prophylaxis and adverse effects thereof is an open question. The fact that the treatment may be worse than the disease in some cases should also be considered.

9. Table VI-II, page 40 indicates up to 5% mortality at 165 rad (250 R) and a cutoff around 100 rad (150 R). Uncertainties in population response suggest that there must be a range around these values and that effects at lower exposure levels are possible.

10. Page 47, Section 6.6.4.2 - There is some confusion about the data studied by the BEIR Committee. Probably most of the data is on

relatively acute exposure to low LET radiation, the type most applicable to the emergency situation studied in the subject report.

11. In Section 6.8 of Appendix VI, the last sentence on page 67 implies that a Monte-Carlo type of determination was employed, as contrasted to the assertion in the second paragraph on page 3.

12. The title to figure VI-8 on page 76 should be changed since the thyroid nodules do not include all thyroid consequences to be expected.

Appendix VII

Editorial

1. Page C-2, item 2 - The core, taken as a whole, cannot "heatup" from sensible heat as stated here.

2. Page C-9 - The "Little Mamu" program should be referenced to supporting documentation.

3. Page I-2, equation (3) - Since this equation involves an integration over time, a distinction in the various time parameters is required since C_g is a function of "t".

Appendix VIII

Clarifications

1. Page A-3, 1st paragraph under Fission-Product Release - It appears that the pin rupture temperature was assumed to be 1500°F in the BOIL code calculations. This does not correspond to either of the two temperatures cited in Appendix VII.

2. Page A-12, last sentence under Bottom Flooding - The meaning of this sentence is not clear, particularly the reference to "these" flooding rates, and the reasoning that heatup of cores at elevated temperatures is not prevented.

Editorial

1. Page 6, top of page - Nomenclature problem: The ECR system described here appears to be the same as the LPRS system used in most of the rest of the Study documentation.

2. Page 7, last sentence - The starting time for CSIS is important. The fact that it must operate for a considerable length of time has nothing to do with start time considerations.

3. Page 8, 1st paragraph under Core Meltdown - It is not clear what is included in SIS failure (not previously defined).

4. Page 34, Accident Time Scale - A discussion similar to this for the PWR case would clarify the PWR containment discussion.

5. Page A-1, 1st paragraph under Core Heatup Calculations - In view of the application of the core heatup results to other PWRs and BWRs, the statement that some of the results apply only to the specific designs considered needs elaboration.

6. Page A-6, equation (A-9) - Q_{MELT} apparently should be Q_{QUENCH}.

7. Page E-9 - The pressures in this assessment should be labelled psig or psia, whichever is appropriate.

Appendix X

Clarifications

1. Page 6, first paragraph - Although the site geology is described, a description of what the plant is actually built on is not mentioned, as was done for the BWR on page 7.
2. Page 45, Note (4) - The Bijlaard formulae have not been defined in the text.
3. Page 94, first paragraph - It is indicated that the LHSIS (LPIS elsewhere) injects into the RCS hot legs. Figure II-53 of Appendix II, Vol. 2, shows injection into the cold legs and the text associated with the figure also indicates cold leg injection.
4. Page 94, third paragraph - The discharge pressure of 300 psig does not appear compatible with the 225-foot head stated on page 275 of Appendix II, Vol. 2.
5. Page 168, item 2 at bottom of page - This item states that the assumption of a 40° tilt of the MSIV actuator axis is a conservative assumption since "one expects vertical installation to be the usual practice." It is not clear why the actual orientation for the Surry plant was not determined in order to establish the validity of this conservatism. (Figure 28 shows an MSIV with about a 40° tilt to the actuator).

Editorial

1. The nomenclature used for the various reactor systems is not consistent with the rest of the Study. Examples are:

App A, Page 18 - Low Head Safety Injection System vs Low Pressure Injection Systems

High Head Safety Injection System vs High Pressure Injection Systems plus Accumulator Systems

Containment Recirculation Spray Systems vs Containment Spray Recirculation Systems

Core Spray Systems vs Core Spray Injection System

Residual Heat Removal Systems vs Post Accident Heat Removal.

Additional Comments

Summary Report

Editorial

1. Page 2, 1st sentence - The sources for the results in Figures 1, 2, & 3 should be identified, and the figures explained in more detail (ie, time period covered, population covered, etc).

2. Page 8, 1st paragraph, last sentence - Depending on schedules and definitions, this statement may be incorrect. Fort St. Vrain (330 MWe-HTGR) should start up this year, and Fulton 1 (1140 MWe-HTGR) is scheduled for startup in 1979.

3. Page 26, Section 2.21, 1st paragraph - A more effective qualification of the WASH-740 results would be to quote the cover letter transmitting the Study to the JCAE in March 1959. This letter, presumably written by the authors of the report, says, in part:

"Pessimistic values, leading to great hazards, were chosen for the numerical values of many uncertain factors which influence the final magnitude of the resulting damage. It can therefore be concluded that these theoretical estimates are greater than the damages which would actually result in the unlikely event of such an accident."

CONTRACT WITH INTERMOUNTAIN TECHNOLOGIES, INC.

CONTINUING WASH-1400 REVIEW TASKS

A. Failure Mode Paths Selected for Review

1.	BWR-Reactor Protection System	Review to determine credit taken for backup Boron injection under BWR transients selected following investigation of BWR Reactor Protection System.
2.	BWR-Transient #1	
3.	BWR-Transient #2	
4.	BWR-Transient #3	
5.	PWR-Electric Power Systems	Independent evaluation of Electric Power System Availability-
6.	PWR-High Pressure Injection System	Review to determine the extent that possible troublesome break locations have been accounted for.
7.	PWR-Small Break #1	
8.	PWR-Small Break #2	
9.	PWR-Loss of Power Transient	Review relationships considered between this accident sequence and specific containment failure modes.
10.	PWR-Low Pressure Injection System	Review to determine range of applicability of assured failure paths and sensitivity of results on accident risk magnitude.
11.	PWR-Low Pressure Recirculation System	

B. Critical Radiological Source Term Parameters Selected for Review

1. Core conditions prior to meltdown calculation
    - a. vessel residual water
    - b. blowdown heat-transfer
    - c. blowdown duration
2. Core meltdown calculation
3. Containment response
    - a. failure pressure
    - b. containment safeguards and containment pressure response.

C-35

PART 2

AUGUST 15, 1975 COMMENT LETTER TO THE NRC

I - I

UNITED STATES ENVIRONMENTAL PROTECTION AGENCY
WASHINGTON, D.C. 20460

15 AUG 1975

Mr. Saul Levine
Deputy Director
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Levine:

The Environmental Protection Agency's comments from the second phase of its review of draft WASH-1400 ("Reactor Safety Study-An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants") are transmitted in this letter.

Because of the significance of the Reactor Safety Study toward establishing the accident risk associated with nuclear power plants, we chose to review the draft report of the study in two phases. The comments from our first phase review, an overall review of the draft WASH-1400, were transmitted to you by our letter of November 27, 1974. The second phase review was an intensive examination of selected areas of draft WASH-1400 to determine if there were deficiencies in their evaluations and to estimate the significance of the deficiencies with respect to the related risk calculations in draft WASH-1400. This effort provided a deeper appreciation of the degree of thoroughness with which the Reactor Safety Study staff has applied the study methodology and of the sensitivity of the study results to changes in individual parameters or in single event probabilities.

We have endeavored to keep your staff informed of the preliminary findings of our second phase review as they became available. It is hoped that the Reactor Safety Study staff finds the comments we are now transmitting, as well as our previous comments, to be useful in their preparation of the final study report and in their refinement of the study methodology.

EPA obtained additional technical support for the review effort through a contract with Intermountain Technologies, Inc. (ITI) of Idaho Falls, Idaho. ITI's report titled "A Review of the Draft Report Reactor Safety Study (WASH-1400)" is transmitted with this letter. This report details ITI's findings from its in-depth examination of selected aspects of the work presented in draft WASH-1400 in support of our second phase review and it serves as documentation supporting some of our comments. At present, only a limited number of copies of the ITI review are available; the report will be issued for general distribution within a few weeks as an EPA report.

While EPA endorses specifically the recommendations of ITI's report and the report's conclusions and observations in general, the following comments emphasize EPA's position, which is based in part on ITI's recommendations as well as on other EPA findings.

The second phase review findings indicate that although errors, omissions and other deficiencies were found in areas of draft WASH-1400, the vast majority of these were found not to have a significant effect on the overall risk estimates. More than a dozen areas were investigated in this phase but the only one which was found to have a significant potential for increasing the estimate of overall risks was the assessment of transient-without-scram accidents for boiling water reactors. The results of our second phase review have not altered our opinion that the Reactor Safety Study provides a forward step in risk assessment of nuclear power reactors, and that the study's general methodology appears to provide a systematized basis for obtaining useful assessments of the accident risks where empirical or historical data are presently unavailable. There are a number of areas of nuclear power technology which should be considered as candidate areas for future application of a refined form of the Reactor Safety Study methodology, including different versions of contemporary light water reactors, high temperature gas cooled reactors, liquid metal fast breeder reactors, and variations such as barge mounted power plants.

The Reactor Safety Study has also served to provide a picture of the state of knowledge of the physical processes and the event sequences that might occur in a nuclear power plant under severe accident conditions and of the consequences of such accidents. It is certain to help in the development of the reactor safety research program, and it may provide insight leading to innovations in reactor protection systems and engineered safety systems.

Although the draft Reactor Safety Study report does not make an absolute judgment on nuclear power plant accident risk acceptability, the comparative risk approach presented in the summary and in the main volume of the draft report is likely to imply an acceptability judgment to the average reader. EPA recognizes that the comparative risk approach is a first step in addressing this question, but by itself is misleading. The summary presentations in draft WASH-1400 serve to illustrate some of the problems with the comparative risk approach, as do some of the observations on the subject in ITI's report. It is not an accurate comparison to compare risks estimated from calculations to risks estimated from experience, to omit latent deaths from comparisons of fatalities nor to compare acute fatalities to latent. A better appreciation of the risk estimates could be gained if their uncertainties were added to the graphs. It should also be acknowledged that the risk from nuclear power is not only the risk from severe accidents, but it also includes the risks from normal operation of nuclear power plants, from associated transportation and storage of radioactive material, from other fuel cycle facilities, and from such potential activities as sabotage and terrorist diversion of materials. It should be made clear in the final WASH-1400 that the study attempts to quantify the risk of accidents from contemporary light-water reactors and does not, by itself, make judgments on the acceptability of quantifications made, although such quantifications may be put into perspective through appropriate comparison with other risks.

Draft WASH-1400 shows that the transient-without-scram accident sequences for boiling water reactors (BWRs) make a major contribution to the overall accident risk. The treatment of several aspects of transient-without-scram accidents should be carried out in more detail to avoid unrealistically high risk estimates; an example is the determination of the combinations of control rods whose failure results in failure to scram. Other aspects of transient-without-scram accidents need better justification of the failure probability values chosen; the assessments of the single control rod insertion failure rate, of the multiple and common mode control rod insertion failure rate, and of the protection provided by the liquid poison injection system are such that higher failure probability values could have been selected from the information given, with a potential for increasing overall risks by as much as a factor of 2.

Some areas were found which appear to be improperly or incompletely considered but for which insufficient information is available to determine quantitatively their risk impact. These areas include human reliability, common mode failure quantification, some aspects of design adequacy, and the techniques for calculating the results of small pipe breaks in pressurized water reactors.

The area of common mode failure, in particular, needs further elaboration, especially because the concept employed in the Reactor Safety Study seems to be broader and inclusive of a greater variety of failures than the usual interpretation of the term. The assertion that common mode failures do not contribute much to the overall risk needs extensive and substantial additional support in the form of comprehensive, logical, and well-connected coverage of the subject. The recent fire at the Browns Ferry plant, an example of a common mode failure which disabled a number of systems of two power reactors simultaneously, emphasizes the need for thorough examination of common mode failure.

The discussion of design adequacy needs to be expanded to include explicit description of the manner in which possible design inadequacies in components, structures, and systems are accounted for in the study methodology.

The core meltdown and containment response analyses in the draft WASH-1400 were found to contain many oversimplifying assumptions. Even though these assumptions may not have a significant effect on the overall risk analysis, better justification for their selection should be provided. These oversimplifying assumptions appear to fall into two classes; assumptions made only for calculational convenience which should be justified by suitable explanation, and assumptions made to bridge over inadequacies in the state of knowledge of physical processes, which should be identified as such to emphasize the need for further research. It appears that there are especially large uncertainties in knowledge of the behavior of the core and its surroundings once the core melting begins. The significance of the oversimplifying assumptions appears to be due to their influence on the probable sequence of events, i.e., whether the heating of the core is so rapid that it melts before effective cooling is restored, and, if effective cooling is not restored, whether the containment fails by excessive internal pressure or by some other mode. For example, in part of the containment failure analysis in draft WASH-1400, it is assumed that a molten core will generate considerable carbon dioxide gas by decomposition of foundation concrete containing limestone aggregate. The analysis of some possible accident sequences shows this gas providing sufficient additional internal pressure to fail the containment before the pressure is relieved into the ground by the molten core penetrating the foundation. The assumption that all foundations contain gas-generating aggregate appears to lead unrealistically to higher risk estimates.

It would seem reasonable from the explanation in draft WASH-1400 of the basis for selection of the pressure at which the containment of the example pressurized water reactor is assumed to fail under accident-created conditions to have selected a lower pressure. This explanation should be expanded to provide more justification for the high pressure selected, because in a number of possible accident sequences the failure pressure appears to be a determining factor relative to release of radioactivity to the atmosphere through the failed containment wall or release into the ground by the core melting through the foundation.

The draft WASH-1400 has also served to call attention to problems associated with the response to an accident to mitigate the consequences to the public. In dealing with an accidental release, the evacuation model of draft WASH-1400 includes a warning time for evacuation which apparently begins at the time of awareness of impending core melt. In order to show that the warning time for evacuation is determined on a practical basis, the final report should give examples of the limiting conditions in the plant which are postulated as bases for the decision to warn the neighboring population to evacuate, and the plant instrumentation indications that will tell the operator that the limiting conditions have been reached.

Examination of design differences between the example pressurized water reactor (PWR) of the Reactor Safety Study and a PWR more representative of the expected 1980 population of PWRs indicates that some results from the example PWR are not applicable to the whole 1980 PWR population. However, the differences in overall risks between contemporary types of PWRs or between contemporary types of boiling water reactors (BWRs) are thought likely to be smaller than those found in the study between the example PWR and the example BWR.'

The enclosed ITI report contains figures showing revisions of health effects data of draft WASH-1400. Each of these figures shows the health effects of draft WASH-1400 adjusted for one suggested change only. These figures do not represent EPA's estimates of consequences because the figures do not include all the necessary corrections to the data of draft WASH-1400. Similarly, the estimates in the discussion of tritium release considerations in ITI's report are sufficient to show that the potential effects of tritium releases are small compared to those from some other types of radionuclides; they should not, however, be construed as presenting an EPA dose model or health effects calculational procedure.

The U.S. Atomic Energy Commission published an "Interim General Statement of Policy" dated August 21, 1974, with respect to the Reactor Safety Study, that states "The study when completed will be the subject of thorough evaluation by the Commission, the independent Advisory Committee on Reactor Safeguards, and the Commission's staff with respect to both the basic question whether the risks portrayed by the study are acceptable from the standpoint of the Commission's statutory responsibility to protect the health and safety of the public, and the related question whether any changes in the Commission's safety or environmental regulations are warranted." A basic conclusion to such an evaluation is whether or not the risks have been portrayed by the study with adequate accuracy and precision. It is recommended that additional care be given to assuring that the results of the study are realistic and to reducing the uncertainties.

Our major reservation with respect to this study is the implied acceptability of the estimated risks to society. Although the study has made major inroads into quantification of accidental risks from nuclear reactors, the acceptability to society of such accidental risks has not been analyzed. It appears that WASH-1400 cannot, nor should it, address the acceptability to society of the risk estimates derived. It is important, however, that WASH-1400 not be susceptible to the interpretation that it presumes such acceptability. Thus, the quantification of risk determined by this study and implications of their acceptability should be clearly differentiated to eliminate any potential confusion. The Reactor Safety Study's summary presentation should be modified to qualify the risk comparisons with more emphasis that they are only a first step toward the evaluation of risk acceptability and that conclusions with regard to the acceptability of the risks can only be drawn when other factors are considered.

We are looking forward to the final report WASH-1400, and we are interested in the plans of the Nuclear Regulatory Commission for further application of the methodology developed in the Reactor Safety Study. In this respect, we urge that a continuous effort be maintained to refine, improve and extend the methodology.

We would be pleased to discuss our comments with you if they require any clarification.

Sincerely yours,

W. D. Rowe, Ph.D.
Deputy Assistant Administrator
for Radiation Programs (AW-553)

Enclosure

PART 3

FINAL REPORT BY ITI

A REVIEW OF THE DRAFT REPORT
REACTOR SAFETY STUDY
WASH-1400

The contract report reproduced as Part 3 of this report was prepared as an account of work sponsored by the Environmental Protection Agency. The contract report is being published so that it will be available as a resource to the scientific community and the general public. It does not necessarily represent the views or policies of the Environmental Protection Agency. In particular, the doses and health effects indicated in the contractor report do not represent EPA's estimates of consequences because the figures do not include all the necessary corrections to the data of WASH-1400.

A REVIEW OF THE DRAFT REPORT
REACTOR SAFETY STUDY (WASH-1400)

By

P. R. Davis

Contract No. 68-01-2244

Project Officer

Dr. Jerry Swift
Office of Radiation Programs

Prepared For
Office of Radiation Programs
U. S. Environmental Protection Agency
Washington, D.C. 20460

2-2

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ABSTRACT

This report presents the results of a review of the draft document, "Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants" (WASH-1400), prepared by the United States Atomic Energy Commission and issued August 1974. The purpose of the review was to provide the Environmental Protection Agency with technical support for assessing the range of applicability of the methods, techniques, and data utilized in WASH-1400. The review consisted of: (1) a selection, based on a preliminary review, of areas in WASH-1400 which appeared to contain errors which could have a significant effect on the results, and (2) an in-depth technical review of each area selected to determine the applicability of the results to the assessment of nuclear power plant risks.

In the sections of WASH-1400 reviewed, certain errors, omissions, and inconsistencies were found. Most of these deficiencies were found not to have a significant effect on the overall results and conclusions of the study, although most tended to increase the calculated risks. In some cases, the effects of an apparent deficiency could not be determined due to lack of information or resources. In these cases, sensitivity studies would be necessary to determine if the deficiencies are significant. One area, the risk contribution from boiling water reactor anticipated transients, was judged to present a potential for significantly increased risks.

WASH-1400 provides a major contribution to risk assessment related to nuclear power plants. The methods, data base, and analysis used in WASH-1400 form an important foundation for a systematic approach

to help identify technical areas needing additional research and development to further reduce risks from nuclear power.

This report was submitted in fulfillment of Contract Number 68-01-2244 by Intermountain Technologies, Inc., under sponsorship of the Environmental Protection Agency. Work was completed as of May 1975.

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7. Mrs. Helen Brown, Intermountain Technologies, for typing and editing.

I. CONCLUSIONS

The purpose of the effort described in this report was to review a draft of WASH-1400 (Reactor Safety Study) in selected areas to determine the range of applicability of the methods, techniques, and data used in determining the risk of nuclear power. The objective of the review was to probe selected areas of the report to the extent necessary (1) to determine if the particular area was correctly evaluated and, if not, (2) to estimate a measure of the significance of any errors, omissions, etc, found in the risk calculations in WASH-1400. This section presents the conclusions of the review.

WASH-1400 represents a comprehensive and much needed assessment of the risks of nuclear power in the United States. It is by far the most extensive such assessment ever attempted, and represents a significant improvement over previous assessments of reactor safety. It will likely become a principal basis for many decisions regarding the safety of nuclear power for generating electricity in the United States and for guiding future reactor safety research and development.

It should be recognized that in any first effort of the magnitude and complexity of the Reactor Safety Study, errors, omissions, and other deficiencies are certain to occur. The existence of these deficiencies should not be construed as invalidating any of the WASH-1400 results unless such deficiencies can be clearly demonstrated to have a substantial effect on the results. Nevertheless, because of the potential importance of WASH-1400 to future policy decisions, it would appear appropriate to correct to the extent possible even the minor deficiencies which do not appear to have a significant effect on the results.

Some of the conclusions presented in this section are based on numerical revisions of a particular area with new risk values computed for the purpose of establishing the significance of an apparent deficiency. It should be emphasized that these revised risk values are, in most cases, gross approximations provided only to establish some measure of the potential significance and to provide justification for a recommendation that a particular area requires additional analysis. The specific conclusions from the review are as follows:

(1) Although errors, omissions, inconsistencies, and questionable assumptions were found in many areas of WASH-1400, the vast majority of these deficiencies were found not to have a significant effect on the overall risk assessments.

(2) The summary presentations in WASH-1400 for comparing the risks of nuclear power with other man-caused risks are sometimes misleading and incomplete. Factors such as obscuring latent deaths from nuclear power, not illustrating calculational uncertainties in nuclear power risks, and making comparisons of calculated nuclear risks with actual risks from other sources without emphasizing the distinction sufficiently, all tend to undermine the strength of the WASH-1400 conclusions, both expressed and implied.

(3) The WASH-1400 risk assessment from transient without scram accidents for boiling water reactors appears to be the most significant analysis problem found in the report. In this case, a preliminary sensitivity study indicates that re-evaluation of the consequences of this accident may increase the WASH-1400 calculated risks from BWRs.

(4) Several areas were found which appear to be improperly or incompletely considered but for which insufficient information is available to determine quantitatively their risk impact. These areas include

- human reliability
- PWR small break calculational techniques
- common mode failure quantification
- some aspects of design adequacy.

(5) The validity of applying the results of the risk assessment using the Surry reactor, chosen to represent all PWRs in WASH-1400, to the 60 to 70 PWRs expected to be in operation by 1980 needs additional consideration in WASH-1400. In several areas, design differences between the Surry plant and a plant more representative of the 1980 plant population indicate that the Surry results may not apply. Surry represents a PWR design similar to only about 20 percent of the anticipated 1980 PWR population.

(6) The core meltdown and containment response analyses were found to contain many oversimplifying assumptions. Although these assumptions may not have a significant effect on the overall risk analysis, very little justification was included in the analyses for selecting the assumptions.

(7) The basis for selecting the PWR containment failure pressure was found to be deficient, and the failure pressure selected appears to be too high.

II. RECOMMENDATIONS

(1) Although most of the deficiencies found in WASH-1400 did not appear to have a significant effect on the overall quantitative risk assessments, such deficiencies should be repaired, along with the major problems found, for the following reasons: (a) the existence of errors, omissions, inconsistencies and questionable assumptions in the report tends to undermine the confidence gained by the reader in the results, especially since in many cases the significance of such deficiencies cannot be assessed without detailed analysis, (b) as the more significant deficiencies in the report are repaired, the effect of some of the minor problems could be amplified. Changes in reactor design, as well as operating and maintenance characteristics, could shift the emphasis and accentuate the significance of deficiencies which presently appear to be minor.

(2) The final results of the Study, especially those portions pertaining to comparisons between nuclear and other man-made risks,should be revised to clearly and consistently indicate:

(a) That the nuclear risks are calculated while other risks are derived from actuarial data,

(b) The substantial uncertainty associated with the nuclear risk calculations,

(c) The latent death risk from nuclear power plants.

(3) The areas of

human reliability,
common mode failures, and
design adequacy

are not quantified or explained to the extent necessary to determine if proper consideration has been given to them in computing overall risks. Some of these areas appear to be improperly considered. It is recommended that additional information be provided which will clearly indicate in a systematic and consistent manner how these areas were evaluated and how the results were included in the risk assessments.

In addition, selected aspects of the use of the failure data considered in the report should be improved, particularly the assessment of pipe rupture probability. These are specifically identified in Section V of this report.

(4) Additional analysis is recommended to justify and confirm the validity of the simplifying assumptions selected for the core meltdown and containment response analysis contained in Appendix VIII of WASH-1400. A reassessment of the PWR containment failure pressure is also recommended as it appears that too high a value was used. A lower value would increase the risks from PWR accidents.

(5) Efforts should commence immediately to extend the Study to cover:

(a) PWR designs other than that represented by the Surry plant selected for analysis in WASH-1400 - Of the 60 to 70 PWRs expected to be operating by 1980 (stated to be covered by the WASH-1400 results), only about 20 percent are of the Surry design. Some of the plants differ in design considerably from the Surry plant, and no assurance is available that the Surry results apply to these plants as assumed by WASH-1400.

(b) Plants which will commence operation beyond 1980 - The Study considers only the 100 reactors expected to be in operation by 1980. This represents only slightly more than 40 percent of all reactors currently operating, under construction, or on order. In view of the fact that the Study, by the time the final report is issued, will have taken something over three years to complete, it is not too soon to begin an extension of the Study to encompass plants scheduled for operation beyond 1980. An analysis of risks from offshore plants should be considered in such an extension.

(c) Gas-cooled reactors - Although gas-cooled reactors at present represent a relatively small (one operating and six ordered) segment of the total reactor commitment in the United States, their risks need to be quantified to complete the reactor risk assessment. It is possible that the gas-cooled reactor could become a significant part of the reactor population in the future. Also, an early risk assessment could identify problem areas which could be eliminated before large scale operation of these plants commences.

(6) The Study should be continuously maintained. It is likely that the power plants covered by the Study will undergo design, operational, and maintenance and testing changes, some of which may be required by regulatory agencies. These changes should be factored into the Study in a timely manner to determine the effect of such changes on the risk evaluation. As the number of operating reactor-hours increases, the component failure rate and accident frequencies should be monitored and periodically factored into the Study. This would improve the statistical basis in the Study,

and could alter some of the results. Continuous maintenance of the Study would not only sharpen the focus of the quantitative risk assessments, but would also have the potential of promptly identifying problem areas, as well as improving the methods used in the Study.

III. INTRODUCTION

In August 1974, the United States Atomic Energy Commission (AEC) issued a draft document, entitled "Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants" (WASH-1400). The document concludes that risks to the general public from power reactor accidents are substantially less than from other man-made risks and most natural disasters.

It is expected that both the methodology and the conclusions of WASH-1400 will have a major impact on the assessment of the risks to the public from the operation of nuclear power plants. It is thus extremely important that the final version of WASH-1400 be as correct as possible in its completeness, technical analysis, data base, and conclusions. It is recognized that in any new effort of the nature and magnitude of the "Reactor Safety Study," errors, omissions, misinterpretation of data, etc, are certain to occur in spite of the best efforts of the authors. However, it is important that such deficiencies be minimized.

The purpose of the effort described in this report was to review WASH-1400 in selected areas to determine the range of applicability of the methods, techniques and data used in determining the risk of nuclear power. The objective of the review was to probe selected areas of the report to the extent necessary in an attempt (1) to establish if the particular area was correctly evaluated and, if not, (2) to establish a measure of the significance of any errors, omissions, etc, found in the risk calculations in WASH-1400.

In many cases the stated risks were numerically revised to assess the impact of the particular discrepancy found in the WASH-1400 analysis. It should be emphasized that these revisions are, in most cases, gross approximations provided only to establish a measure of the potential significance and to provide justification for a recommendation that a particular area be re-analyzed. In most instances, a rigorous and detailed analysis, beyond the scope of the effort described herein, is required to arrive at a definitive numerical revision of the risk.

The effort proceeded in three subtasks, as follows:

Subtask I - This phase consisted of a general review of the entire document to select those areas requiring an in-depth assessment. This phase culminated in the preparation of a work plan which described these areas and the basis for their selection. This work plan is included in this report as Appendix A. Each area was selected based on (1) a determination that it may have a significant impact on the overall risk and (2) a determination that errors may exist in the analysis of the area as presented in WASH-1400. The areas selected were grouped into two categories. The first category included those areas related to failure mode paths identified in WASH-1400. These failure mode paths included the accident event sequence as well as the failure modes of the safety system designed to control the accident. A total of 12 areas were initially selected for an in-depth assessment. This number was later reduced to 10 as a result of combining and slightly altering two areas. The second category consisted of those areas related to establishing the consequences of each accident sequence. The areas were selected based on: (1) the area should significantly influence either the magnitude or the time of release of the radioactive material from the containment, and (2) the area, based on the Subtask I review, was suspected to be incorrectly quantified, improperly applied,or significant but not considered. A total of six areas were initially selected for analysis. This number was later reduced to four when it was found expedient and logical to combine some of the areas.

Subtask II - This phase consisted of an in-depth assessment of the failure mode paths selected in Subtask I of the WASH-1400 review. The paths initially selected consisted of the following:

Table 1 - INITIAL FAILURE MODE PATH REVIEW AREAS

1. BWR reactor protection system failure
2. BWR anticipated transient accident sequence (#1)
3. BWR anticipated transient accident sequence (#2)
4. BWR anticipated transient accident sequence (#3)
5. PWR electric power system failure
6. PWR high pressure injection system failure
7. PWR small break loss of coolant accident sequence #1
8. PWR small break loss of coolant accident sequence #2
9. PWR loss of power transient accident sequence
10. Component failure rates (BWR and PWR)
11. PWR low pressure injection system failure
12. PWR low pressure recirculation system failure

After starting the review of each of the above areas, it became evident that some modifications to the list were in order. The first such change was made to items 2, 3 and 4 (BWR anticipated transient accident sequences). It became clear that an in-depth analysis of these accident sequences would require resources beyond the limits provided in the contract. Each analysis would require the use of complex computer codes in which transient thermal-hydraulic and reactivity effects must be accounted for. (An example of such a calculation may be found in Reference 1).

Thus, items 2, 3 and 4 in Table 1 were replaced by a single item, entitled "BWR Transient Accidents."

The second modification occurred after preliminary investigation into items 7 and 8 (PWR small break accident sequences #1 and #2) revealed that, in order to properly investigate small break accidents, it would be necessary to perform complex transient thermal-hydraulic calculations. These calculations would require the use of computer codes not available except in proprietary versions used by PWR vendors. In reviewing the capability of such codes, it was found that substantial differences in results existed in the calculations as published by the vendors. It was thus concluded that a useful task would be to perform a detailed review of existing analytical techniques aimed at evaluating the credibility of the results. Since WASH-1400 assumes, based on vendor calculations as approved by AEC-Regulatory (now the Nuclear Regulatory Commission), that adequate core cooling is provided for all small break accidents (if emergency core cooling systems operate as designed), it becomes important to examine the vendor calculations. This effort takes on added significance based on the WASH-1400 conclusion that small break accidents are a dominant contributor to the overall PWR risk assessment. Thus, items 7 and 8 of Table 1 have been replaced by a single item, entitled "PWR small break loss of coolant accident analysis review."

The third, and final, modification was to add a review of the BWR loss of power accident to the list. This item was added when inconsistencies and errors were found in the assumptions relative to operation of the emergency diesel generators in the BWR fault tree of WASH-1400.

As a result of these modifications, the initial list of review areas (Table 1) was changed to the following list:

Table 2 - FINAL FAILURE MODE PATH REVIEW AREAS

1. BWR reactor protection system failure
2. BWR transient accidents
3. BWR electric power system failure
4. PWR electric power system failure
5. PWR high pressure injection system failure
6. PWR small break loss of coolant accident analyses
7. PWR loss of power transient accident sequence
8. BWR-PWR component failure modes and rates
9. PWR low pressure injection system failure
10. PWR low pressure recirculation system failure

The review procedure was identical for the areas in Table 2 which consist of "systems." Thus, for areas 1, 3, 4, 5, 9 and 10, the review consisted of three principal parts. The first part involved a detailed review of the fault trees presented in WASH-1400 (Appendix II, Vol. 2 and 3) to determine if the failure modes and associated fault tree logic were consistent and correct. In addition, failure rates were traced through the trees to assure that the final system unavailability values were correctly computed. The second part consisted of determining if significant potential failure modes of each system could be found which were overlooked. The third part of the procedure (applicable only to PWR systems) consisted of a comparison of the system as analyzed in WASH-1400 for the Surry reactor to a plant more representative of contemporary PWR design.

The reactor selected for this comparison was the Trojan(2) reactor which is scheduled to come on line in 1975. This third part was included since the results in WASH-1400 are alleged to apply to approximately 100 reactors (Summary Report - WASH-1400) which are expected to be in operation by 1980, although the detailed fault tree and consequences analyses of WASH-1400 were specifically done for the Surry and Peach Bottom reactors. The Surry(3) reactor is a 2441 MWt, three-loop pressurized water reactor designed by Westinghouse Electric Corp. Of the approximately 67 pressurized water reactors scheduled to be operating by 1980(4,5), only 14 (21 percent) will be of the three-loop Surry design. Approximately 22 (33 percent) will be of the four-loop design represented by Trojan. (Beyond 1980, based on current reactor orders, the number of four-loop plants is expected to exceed 50 percent of all PWRs.) It was thus concluded that the applicability of the WASH-1400 analyses to all PWRs expected to be operating by 1980 could best be determined by ascertaining the applicability of the results to a representative four-loop PWR. It is also considered appropriate that a similar comparison be made for other PWR designs (see Section II, Recommendations).

In order to accomplish the Surry-Trojan comparisons, a detailed compilation was prepared of the design and operating features for the Trojan systems selected in Table 2. From this compilation, the Surry fault trees from WASH-1400 were examined to determine if they were applicable to the Trojan system. Any differences were noted, and the significance of the differences were evaluated where possible with existing information.

With respect to BWRs, the Peach Bottom design analyzed in WASH-1400 represents 67 percent (22) of the 33 BWR plants scheduled to be operating in 1980. Thus, Peach Bottom is representative of the majority of BWRs to which the WASH-1400 results are stated to apply, and a comparison between Peach Bottom and a BWR of more contemporary design was not deemed necessary as part of this review. It should be noted, however, that the basic Peach Bottom design, particularly the containment structure (designated Mark I) has been superseded by a sequence of two designs called Mark II and Mark III(6). Beyond 1980, the Mark I design will represent an increasingly smaller number of BWR plant designs.

Summarizing, the review of each Table 2 system consisted of: (1) a review of the failure modes presented in WASH-1400 including the application of failure rate data; (2) an investigation to determine if any additional significant failure modes of the system could be found, and (3) a comparison between the PWR system analyzed in WASH-1400 and a PWR reactor of more contemporary design.

For the accident sequences listed in Table 2 (areas 2 and 7), the "event tree" sequence as contained in Appendix I was reviewed along with supporting information contained elsewhere in WASH-1400. The review consisted of determining if the accident event sequence and time of events were properly considered. Where appropriate, the probability of the accident was reviewed. The effectiveness of systems designed to mitigate the accident sequence was explored in some cases, and events considered likely to occur which were not discussed in WASH-1400 were analyzed.

Area 6 of Table 2, "PWR small break loss of coolant accident analyses," has been discussed previously in this section. In summary, this review consisted of determining if the small break loss of coolant accident analysis performed by the vendors is appropriate as used in WASH-1400.

Area 8 of Table 2, "BWR-PWR component failure rates," consisted of reviewing Appendices III (Failure Date), IV (Common Mode Failures), and X (Design Adequacy) of WASH-1400. The review was directed towards determining that:

(a) the data sources utilized in WASH-1400 were applicable and properly applied,

(b) applicable data sources were considered,

(c) common mode failures were properly considered and accounted for,

(d) appropriate attention was given to considerations of design adequacy, and

(e) component failure rate values were properly computed from the data used.

In all of the areas listed in Table 2, miscellaneous errors, inconsistencies, etc, which were judged to have only minor impact on the results, have been included in a separate section of each review.

Subtask III - This subtask consisted of analyzing accident consequence areas selected during Subtask I. The areas selected and the basis for selection are discussed in Appendix A. The areas consisted of those factors and parameters affecting the magnitude and time sequence of release of fission products to the containment from the core during an accident. Excluded from the analysis contained herein, with the exception of tritium release, was any consideration of factors affecting the distribution and biological effect of fission products beyond the containment boundary. A review of this area (Appendix VI of WASH-1400) was done independently by the Environmental Protection Agency.

The initial list of areas selected is shown in Table 3.

Table 3 - INITIAL CONSEQUENCE AREAS SELECTED FOR REVIEW

1. Core Parameters Prior to Meltdown Calculation
2. Core Meltdown Calculations
3. Containment Response - Failure Pressure (PWR)
4. Containment Response - Pressure History (PWR)
5. Applicability of Containment Response to Other BWR-PWR Systems
6. Tritium Release Considerations

This list was modified when it became obvious that continuity could be improved by combining 1 with 2 and 5 with 3. In general, it was impossible to quantitatively determine the effect of different parametric assumptions regarding the state of the core prior to the meltdown (area 1) transient without performing the meltdown calculation (area 2). Area 5 was combined with area 3 in order to provide a more direct comparison of containment response between the reactors selected in WASH-1400 and those of more recent designs. Thus, the consequence areas selected became the following:

Table 4 - FINAL CONSEQUENCE AREAS SELECTED FOR REVIEW

1. Parametric Studies of Core Meltdown (PWR)
2. Containment Response - Failure Pressure (PWR)
3. Containment Response - Pressure History (PWR)
4. Tritium Release Considerations (BWR-PWR)

IV. GENERAL RESULTS

The results of the review of the areas described in Section III and Appendix A are presented in summary form in Tables 5, 6, 7 and 8. Table 5 presents the results of the review of emergency systems designed to control the consequences of the various accidents considered in WASH-1400. Each system is considered, and the results of each part of the review are presented. The results are classified as discrepancies having an insignificant (I), significant (S), or indeterminate (M) effect on the overall risk. The letter "N" is used to indicate that "no discrepancy was found. "Significant" is defined as any change that appears to result in a factor of two or greater change in the acute deaths computed in WASH-1400. The symbol (+) means that the risks would appear to be increased if the discrepancy were resolved and (-) means they would appear to be decreased.

Table 6 is a similar comparison of the two accident sequences reviewed. As in Table 5, the results of each accident sequence review are indicated by symbols in the table. The legend below the table explains the symbols, which are similar to those used in Table 5.

The results of the item 8 (BWR and PWR Component Failure Rates) review area are shown in Table 7. Three areas were evaluated: Component Failure Data (Appendix III), Common Mode Failures (Appendix IV), and Design Adequacy (Appendix X). The main general review categories are shown. "Quantified" refers to the extent to which the information in each Appendix was numerically evaluated in a form suitable for application to the main WASH-1400 risk assessments. "Complete" refers to the extent to which applicable data sources were used and whether appropriate aspects of the subject being evaluated in each Appendix were

Table 5 - RESULTS OF REVIEW OF ENGINEERED SAFETY SYSTEMS FAILURE

		BWR Reactor Protection System	BWR Electric Power System	PWR Electric Power System	PWR High Pressure Injection System	PWR Low Pressure Injection System	PWR Low Pressure Recirculation System
Review Phase	Errors	S (+)	I (+)	I (+)	I (+)	I (+)	I (+)
	Omissions	N	N	I (+)	I (+)	N	I (+)
	Comparison-PWR only	--	--	M (-)	M (+)	M (-)	I (+)

N - no discrepancies were found.

I - discrepancies were found, but were assessed to result in an insignificant change in the overall risk.

M - discrepancies were found, but the risk impact could not be definitely determined due to limited resources or insufficient information.

S - discrepancies were found and were judged to be potentially significant in terms of impact on the overall risk.

(+) - correction of the discrepancy would result in an increase in the overall risk.

(-) - correction of the discrepancy would result in a decrease in the overall risk.

Table 6 - RESULTS OF REVIEW OF ACCIDENT SEQUENCES

	BWR Transient Accident	PWR Loss of Power Transient Accident
Errors	I (-)	N
Omissions	N	I (+)

N - no discrepancy found.

I - discrepancy found, but assessed to result in an insignificant change in the overall risk.

M - discrepancy found, but the risk impact could not be definitely determined due to limited resources or insufficient information.

(+) - correction of the discrepancy would result in an increase in the overall risk.

(-) - correction of the discrepancy would result in a decrease in the overall risk.

Table 7 - RESULTS OF REVIEW OF COMPONENT FAILURES

	Quantified	Complete	Applied	Errors
Component Failure Data	Yes	Yes	Yes	Yes
Common Mode Failures	Partially	Yes	Partially	No
Design Adequacy	No	No	Indeterminate	No

considered. "Applied" refers to the extent to which the results were translated into the main stream of the WASH-1400 risk evaluations. The "Errors" column is an assessment of whether errors of potential significance were found in each Appendix reviewed. In most cases, due to incomplete information, it was not possible to establish the significance of the shortcomings found in each Appendix. The results shown in Table 7 represent general assessments in the four rather broad categories. Specific and detailed assessments can be found under Section V, item 8.

The results of the consequences area review are shown in Table 8. Except in the case of Tritium Release, which was not discussed in WASH-1400, no omissions were found. There were, however, apparent errors in the WASH-1400 analyses for the other three areas, as shown in Table 8. In none of the cases was an attempt made to quantify the change because of limited resources or lack of sufficient information.

Table 8 - RESULT OF REVIEW OF ACCIDENT CONSEQUENCE AREAS

	Parametric Studies of PWR Core Meltdown	PWR Containment Response-Pressure History	PWR Containment Response-Failure Pressure	BWR-PWR Tritium Release
Errors	M (+)	M (?)	M (+)	(1)
Omissions	N	N	N	I (+)

(1) not considered in WASH-1400.

I -discrepancy found but assessed to result in an insignificant change in the overall risk.

M -discrepancy found, but the risk impact could not be definitely determined due to limited resources or insufficient information.

N -no discrepancy found.

(+) -correction of the discrepancy would result in an increase in the overall risk.

(?) -change in risk could not be determined.

V. ANALYSIS

This section presents the detailed analysis of each of the areas selected in WASH-1400 for review as discussed in the Introduction and in the work plan included as Appendix A. The analysis is divided into two parts. The first part includes those areas which affect the failure mode paths considered in WASH-1400. These ten areas are listed in Table 2 of the Introduction, and include: (a) six reactor systems, (b) three accident sequences, and (c) an assessment of component failure modes and rates.

The second part of the analysis includes the four consequence areas selected for review as listed in Table 4 of the Introduction. For these four areas, a review was made of the WASH-1400 analysis (if considered therein), and an independent analysis was conducted and compared with WASH-1400 results.

A. FAILURE MODE PATHS

The ten failure mode paths considered in this part of the analysis were reviewed in the following manner: For the six reactor systems, the review consisted of a general review of information contained in WASH-1400, an assessment of any omissions found, and, for the PWR systems, a determination of the applicability of the Surry reactor system analysis to the corresponding Trojan reactor system. For the three accident sequences, the review consisted generally of an assessment of the accident sequences considered and the validity of selected assumptions. The assessment of component failure modes and rates includes a review of Appendices III (Failure Rate Data), IV (Common Mode Failures), and X (Design Adequacy) of WASH-1400. The review proceeded as follows:

1. BWR Reactor Protection System Failure

General Review - An in-depth review was performed of the BWR reactor protection system failure-to-scram analysis. The analysis, presented in Appendix II, Vol. 3, Section 6.2 of WASH-1400, was found to be correct except for two areas in the analysis of three adjacent rods failing to insert upon scram demand, and the credit taken for liquid poison injection.

The probability that three adjacent rods will fail to scram from hardware failures is assessed in Section 6.2, page 106 of WASH-1400 to be 5.8×10^{-6}. This event is a major contributor to the failure-to-scram probability from all sources (1.3×10^{-5}) derived in WASH-1400. It is assumed that the failure of any three adjacent rods to insert results in failure to render the core subcritical. This assumption is described as "extremely conservative," and it is not clear why a more realistic determination was not attempted, particularly in view of the fact that the charter for the Reactor Safety Study specifically called for a "realistic assessment" (page 15, main document). As will be seen, this assumption can have a substantial influence on the calculated risks to the public from BWRs.

The value of 5.8×10^{-6} was derived, through a series of calculations and manipulations, from the failure rate data derived from BWR operating experience as described in Section 3.2 of Appendix III to WASH-1400. According to Section 3.2, there have been two control rod failures reported out of 16,200 individual rod insertions. This yields a failure rate of $2/16{,}200 = 1.2 \times 10^{-4}$ per demand, which was rounded to 1.0×10^{-4}. This means that, for a core with 185 control rods (representative of current BWRs), a rod insertion failure would be expected to occur once in every 54 scrams. This appears to be a low, but not necessarily unreasonable, number. However, in view of the potential critical importance of this value, it would be appropriate to substantiate its validity. In particular, the data base should be probed to assure that:

(1) All rod insertion failures were indeed reported. There is a tendency not to report failures which are easily correctable, not considered important, or are not reported due to misinterpretation of the AEC's failure reporting criteria.

(2) All applicable data have been included. The data base, two failures out of 16,200 for the year 1972, provides only 90 percent confidence that the failure rate is between $2x10^{-5}$ and $3x10^{-4}$, viewing the 16,200 scrams as a large number of binomial tests. This uncertainty was not considered in Appendix III. More recent data from 1973 to date, mentioned on page 31 of Appendix III, and alleged to support the 1972 data, should be included to statistically improve the failure rate. In addition, Table III-5 shows a total of six BWR control rod failures (which would yield a failure rate of $3x10^{-4}$) not two as indicated on Table III-7 from which the failure rate was derived. Table III-7 is said to be compiled from data in Table III-5, and the number of failures correspond in all cases except control rods. It is not clear why the difference exists.

An extensive fault tree analysis of a BWR control rod insertion failure during scram was performed by Acero(7). He concludes, based partially on data obtained from General Electric Co., that the probability of a control rod failing to insert upon a scram demand is about $3x10^{-3}$, a factor of 30 higher than the value used in WASH-1400. While this value seems unreasonably high and not supported by reported experience (also, some assumptions in Acero's work can be disputed), the difference needs to be resolved.

To determine the sensitivity of the calculated overall public risk from BWRs to the failure probability of control rod insertion upon scram demand, a sensitivity study was performed. Table 9 gives the increase in probability of a radioactive release in BWR release

Categories 1, 2 and 3 (Appendix V, page 38), assuming a value of $1\text{x}10^{-3}$ for the control rod insertion failure probability.

The first column, "RPS Failure Probability," is the failure probability of the reactor protection system from all causes, including failure to insert three adjacent control rods. "Total Scram Failure Probability" includes the RPS failure plus manual initiation of the liquid poison injection system.

It is emphasized that technical justification does not exist for using a failure probability of 10^{-3} for a single control rod insertion failure. It is simply a convenient value with which to establish sensitivity, and it lies between the WASH-1400 number and Acero's result. Categories 1, 2, and 3 are the only categories out of the six included in WASH-1400 which are affected by the failure to scram.

Table 9

SENSITIVITY OF BWR RELEASE PROBABILITY TO CONTROL ROD FAILURE PROBABILITY

	RPS Failure Probability	Total Scram Fail. Prob.	Release Probability		
			Categ.1	Categ.2	Categ.3
WASH-1400 evaluation with single control rod failure = $1\text{x}10^{-4}$ (1)	$1.3\text{x}10^{-5}$ (2)	$4\text{x}10^{-7}$ (2)	$9\text{x}10^{-7}$(3)	$2\text{x}10^{-6}$(3)	$1\text{x}10^{-5}$(3)
Sensitivity study evaluation with single control rod failure = $1\text{x}10^{-3}$	$3.9\text{x}10^{-4}$	$1.2\text{x}10^{-5}$	$7\text{x}10^{-6}$	$2\text{x}10^{-5}$	$9.8\text{x}10^{-5}$

(1) Appendix III, Pg. 39, WASH-1400

(2) Appendix V, Pg. 68, WASH-1400

(3) Appendix V, Pg. 38, WASH-1400

Table 10 gives the change in overall risk from BWR accidents, in average acute fatalities per year, as a result of changing the single control rod failure probability from 10^{-4} to 10^{-3}.

Table 10

INCREASE IN AVERAGE ACUTE FATALITIES PER YEAR FROM INCREASE IN CONTROL ROD FAILURE PROBABILITY(4)

	Category	Probability Per Year	Average Acute Fatalities	Average Acute Fatalities/yr
WASH-1400	1	$9x10^{-7}$	1.7	$1.5x10^{-6}$
	2	$2x10^{-6}$	48.0	$9.6x10^{-5}$
	3	$1x10^{-5}$	3.0	$3x10^{-5}$
	4	$3x10^{-5}$	3.9	$1.2x10^{-4}$
	5	$1x10^{-5}$	1.1	$1x10^{-5}$
			WASH-1400 TOTAL =	$2.7x10^{-4}$
SENSITIVITY STUDY	1	$7x10^{-6}$	1.7	$1.2x10^{-5}$
	2	$2x10^{-5}$	48.0	$9.6x10^{-4}$
	3	$9.7x10^{-5}$	3.0	$2.9x10^{-4}$
	4	$3x10^{-5}$	3.9	$1.2x10^{-4}$
	5	$1x10^{-5}$	1.1	$1x10^{-5}$
			SENSITIVITY STUDY TOTAL =	$1.4x10^{-3}$

(4) All WASH-1400 values from Table VI-20, Appendix VI, pg. 71.

Thus, changing the single control rod failure probability from 10^{-4} to 10^{-3} results in an increase of about a factor of five in the calculated average acute fatalities per year from all BWR accidents.

Since the BWR risks, based on the foregoing analysis, appear to be quite sensitive to the probability of a single rod failing to scram upon demand during a transient accident, it is important that the single rod scram failure probability be accurately assessed. In particular, additional, more extensive data (which is apparently available) should be included in the assessment; the reason for including only two of six reported failures needs to be analyzed and explained; and Acero's analysis should be considered.

A second area which appears somewhat questionable in the WASH-1400 analysis of RPS failure occurs on page 105 of Appendix II (Vol. 3). In determining the probability of three adjacent control rods failing to insert on scram demand, consideration is given to common mode failures. On page 105, the following assessment is present:

> "The probability of any three rods failing to enter the core was assessed on the basis that complete independence is a nonconservative assumption, while tight coupling of these failures is probably overly conservative. Complete independence yields:
>
> $$(1\times10^{-4})(1\times10^{-4})(1\times10^{-4}) = 1\times10^{-12}$$
>
> A tight coupling assessment is based on observed common mode failures, which are approximately 10 percent of observed failures. However, the majority of these cause only degradation of the component, approximately 10 percent result in failure. Thus, the assumption of tight coupling yields:
>
> $$(1\times10^{-4})(1\times10^{-2}) = 1\times10^{-6}$$
>
> The log-normal median between these values is:
>
> $$(1\times10^{-12})(1\times10^{-6}) = 1\times10^{-9}$$
>
> Using this value..."

The value of 1×10^{-9} was subsequently used to compute the RPS failure probability. The above discussion in WASH-1400 seems to imply that the common mode contribution is 0.01 times the single component failure rate. Thus, the actual value to be used, based on this discussion, would appear to be 1×10^{-6}, rather than some combination (in this case, log-normal median) with the uncoupled failure rate.

Obviously, the "tight coupling" assessment of 0.01 times the single rod failure rate is not based on observed common mode failures of control rods since too few have occurred. However, the "observed failures" from which the 0.01 was derived are not identified, nor is there any justification for applying this rate (and assuming it represents a "tight coupling") to BWR control rod failures. Both of these considerations must be included in order to establish the validity of the assessment. In addition, the basis for combining a common mode failure contribution with a loose coupling failure contribution by use of a log-normal median technique needs explanation and justification.

Based on the foregoing discussion, a computation was done assuming a value of 1×10^{-6} for three adjacent rods failing to insert. The net result was to raise the calculated total BWR risks by a factor of 30 and the average risks correspondingly (see Figure 1.)

A third consideration relative to the probability of reactor shutdown which appears to be incorrectly assessed in WASH-1400 is the credit taken for operator action in activating the liquid poison injection system. According to Appendix V (page 68) of WASH-1400, the failure of "Reserve Shutdown," which includes automatic recirculation pump trip plus manual actuation by the operator of the liquid poison injection system, is dominated by operator failure and is assessed to be 3×10^{-2}. It does not appear reasonable to take credit for manual actuation of the liquid poison injection system in the event of RPS failure for the following reasons:

(1) According to the description in Reference 8, in order to actuate the liquid poison injection system, the operator must locate a key, insert it in the proper console location and turn it. This action fires explosive valves which allow the injection of sodium pentaborate solution into the primary system. Since BWRs are designed to operate without any soluble poison in the primary system, the injection of sodium pentaborate solution is an undesirable

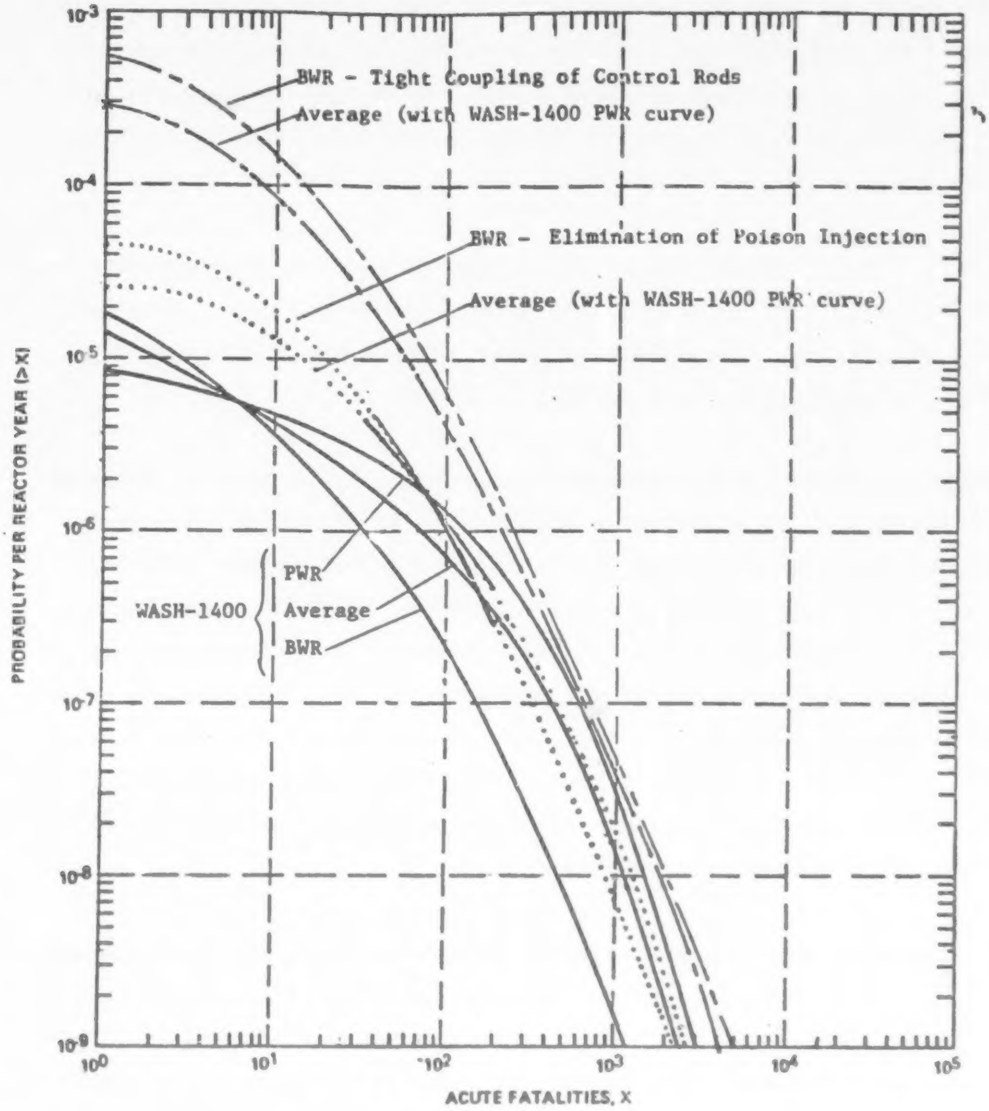


Figure 1 - Effect of Changes in BWR Reactor Protection System Analysis

event, requiring extensive system cleanup. The operator is undoubtedly aware of these consequences from liquid poison injection, and will be inhibited from using the system. The use of a key to actuate the system imposes additional requirements on the operator over the use of a simple switch. Such requirements, coming during a high stress condition caused by failure to scram during a reactor transient, should increase substantially the likelihood of operator error. Table III-13 in Appendix III of WASH-1400 assesses the error rate of an operator after the first 30 minutes in an extreme stress situation as 10^{-1}, and even greater for shorter times. In order for the liquid poison injection to be effective, the operator must fire the explosive valves within 10 minutes of the transient event(9,10). In view of these considerations, it is not clear why an operator error rate of $3x10^{-2}$ was used in WASH-1400.

(2) Recent studies of anticipated-transient-without-scram accidents have concluded that manual injection by the operator of liquid poison cannot be considered, in most cases, as a backup to the reactor protection (scram) system. WASH-1270(11) concludes, on page 33, "Liquid poison injection systems (LPIS), as designed and used at present, have too slow a response to deal effectively with some possible failure-to-scram circumstances." The report subsequently discards manual poison injection as a backup for scram failure during anticipated transients. A report by General Electric(12), manufacturer of BWRs, concludes "This analysis (of anticipated transients) indicated that manual action (of the liquid poison injection system) would be too slow to be generally applied, and automatic initiation of the liquid control system is required."

(3) Aside from the issue of operator reliability, there is some question regarding the ability of the liquid poison injection to effectively reduce the core power level quickly enough to prevent

> core damage in the event of an anticipated transient. General Electric(12), in response to an AEC-Regulatory requirement(11) that BWR reactor protection system reliability during anticipated transients be augmented, has proposed, in addition to automatic initiation of the poison injection system, an increase in the "liquid (poison) control system capacity." This proposal is currently under review by the Nuclear Regulatory Commission. It thus appears, according to General Electric, that even if the liquid poison is injected automatically immediately after the anticipated transient, its capacity may presently be insufficient to control the event as evaluated against NRC requirements.

In view of the foregoing considerations, it does not appear reasonable to consider manual actuation of the liquid poison injection system as a backup to the reactor protection system in the event of some severe anticipated transients. To assess the effect of eliminating the LPIS from consideration, a sensitivity computation, similar to that performed for the reactor protection system, was completed. A value of 1.3×10^{-5} (WASH-1400 evaluation of RPS failure) was used for failure of reactor shutdown instead of 4×10^{-7} (used in WASH-1400 considering liquid poison injection). The results are shown in Figure 1, and are about the same as for the factor of 10 increase in control rod insertion failure probability analyzed earlier in this section. The increase in probability results in about a factor of 5 at a given acute fatality number. The average risk curve would increase by about a factor of 3 at low acute fatalities, and a negligible amount at high acute fatalities. However, this is a conservative assessment since it does not appear that all 10 of the anticipated transients assumed to occur per year in WASH-1400 are severe enough to create an offsite risk in the event of scram failure. As discussed in item 2 of this section (BWR Transient Accidents), only about three severe transients per year might be anticipated. This would nearly offset the increase in risk associated with eliminating credit taken for manual liquid poison injection discussed previously. Since a wide variety of initiating events are lumped into anticipated transient accidents, it would seem prudent for the WASH-1400 analysis to consider these separately to determine under what circumstances scram is required, how quickly core shutdown is needed to prevent core melt, and under what conditions the liquid poison injection system can be considered as a backup. The WASH-1400 analysis presently considers all anticipated transients on the same bases.

It should be noted that the changes proposed by General Electric in Reference 12 could, if accepted by the AEC and shown to be effective, reduce the probability of reactor shutdown failure in the event of an anticipated transient. Since WASH-1270 recommends an increase in reactor shutdown reliability, some modifications to the BWRs, if not GE's current proposal, will presumably be made. These modifications will alter the WASH-1400 risk assessment.

Omissions - No omissions were found in the WASH-1400 analysis.

Conclusions - The effect of the revisions to WASH-1400 considered in this section would result in an increase in BWR risks. The quantitative increases presented serve only to illustrate the sensitivity of questionable analyses in WASH-1400 and the need for additional, study. Anticipated future changes in BWR reactor shutdown reliability would be expected to reduce the BWR risks. A realistic assessment of the number and location of control rods required to shut down the reactor would also likely lead to reduced risks, as indicated at the beginning of this section.

Miscellaneous Comments - The following comments on WASH-1400 were developed during the course of the review of the BWR Reactor Protection System:

(1) Appendix II, Vol. 3, Page 105-106 - The entire development of the hardware contribution to the reactor protection system failure

(Q_H) needs substantially more description and explanation. The first sentence of page 105 should, it appears, read "The probability of any specific three rods..." Also, the numerical value of $P(3A) = 3.0\times10^{-6}$ cannot be obtained from the preceding equation. Also, it appears in the analysis that random (rather than adjacent) rod failures sufficient to cause scram failure have not been considered, nor have configurations of greater than three adjacent rods.

(2) Main Document, Page 236 - The statement that there have been 2000 reactor years of military and commercial power reactor experience with no nuclear accidents is not correct. Depending on the definition of "nuclear accidents," numerous accidents have occurred and, in at least one case, a stationary, low power, military reactor (SL-1) experienced a nuclear excursion which killed three people. (Due to differences in design, the initiating event for this accident would not cause an excursion in commercial power reactors.)

(3) Appendix II, Vol. 3, Section 87 - It is not clear why the values computed for RPS unavailability differ so much throughout this section. The median unavailability computed by Monte Carlo simulation is quoted as 1.3×10^{-5} (Q_{MED}) on page 92. The point estimates which contribute to Q_{MED} add up to 8×10^{-6}, and the Contribution Pictorial Summary (Fig. II-131, page 123) lists a value of 2.47×10^{-6} as the total for RPS unavailability. Also, the triple failure contribution in this figure (2.8×10^{-7}) doesn't agree with the point estimate (5.8×10^{-6}) on page 92.

(4) Appendix V, Page 68 - The RPS failure value in the figure was obtained from the value calculated on page 92 of Appendix II, Vol.3. However, this latter value was based on the occurrence of a LOCA, while the application on page 68, Appendix V, is for transient

events. Although a difference in RPS failure probability may not be significant between a Loss-of-Coolant-Acident (LOCA) and a transient, different sensors measuring different quantities are used to initiate scram for the two accidents.

2. BWR Transient Accidents

General Review - BWR transient accidents are described and analyzed in Appendix I, Section 4.3.2 of WASH-1400. The accidents appear to be properly considered except for the assumptions made regarding the likelihood of the initiating event.

Likelihood of transient event - The assumed frequency in WASH-1400 for "anticipated" transients is 10 per year, as assessed on pages 56 and 57 of Appendix V, WASH-1400. This frequency is based on information contained in Reference 13, which indicates about 10 BWR transients per reactor year. However, only about three of these shutdowns were severe enough to require an immediate core shutdown to prevent core damage. That is, only three would qualify as anticipated transients requiring quick core shutdown. During 1973, according to Reference 14, an average of two such transients occurred per reactor. Reference 12 develops a list of nine transients, based on operational experience, which "have the potential of a frequency of occurrence of at least once in four years of reactor operation at power conditions such that a significant transient results and scram is called upon to shut down the reactor." Assuming, on the average, nine transients in four years per reactor results in 2.25 transients per reactor year. It thus appears that a realistic estimate of anticipated transient frequency would be three per year rather than 10 as used in WASH-1400.

A factor of three reduction (from 10 to 3) in the yearly frequency of anticipated transients would reduce the probability of release by a factor of three for BWR release categories 1 through 4, since each is dominated by anticipated transient accidents. This would, in turn, reduce the probability-acute fatality curve by about a factor of three as shown in Figure 2. The average curve would be reduced by a maximum factor of two at the low end of the acute fatality scale, also as shown on Figure 2.

Omissions - No omissions were found in the treatment of anticipated transient accidents in WASH-1400.

Conclusions - Based on operating experience, the frequency of BWR anticipated transients should be reduced from 10 to 3 per year per reactor. This would result in a reduction in calculated BWR risks as well as in average risks, as shown in Figure 2. As discussed in the preceding item of this section (BWR Reactor Protection System Failure), these transients should be considered separately depending on severity and core power shutdown requirements.

Miscellaneous Comments - The following comments on WASH-1400 were developed during the course of the review of the BWR Transient Accident:

(1) Appendix I, Page 207 - Footnote 1 states that either the scram system or the combination of pump trip and (manually operated) soluble poison injection are "sufficient for the very likely and less likely transients." This statement implies that these transients are slow enough that manual operation of the poison injection system can be assumed as an effective shutdown mechanism. As described under item 1 in this section, this assumption is questionable.

3. BWR Electric Power System Failure

General Review - The BWR Loss of Power analysis presented on pages 9-81, Appendix II, Vol. 3 of WASH-1400 was evaluated. The system appears to be properly analyzed except for the assumption made regarding the

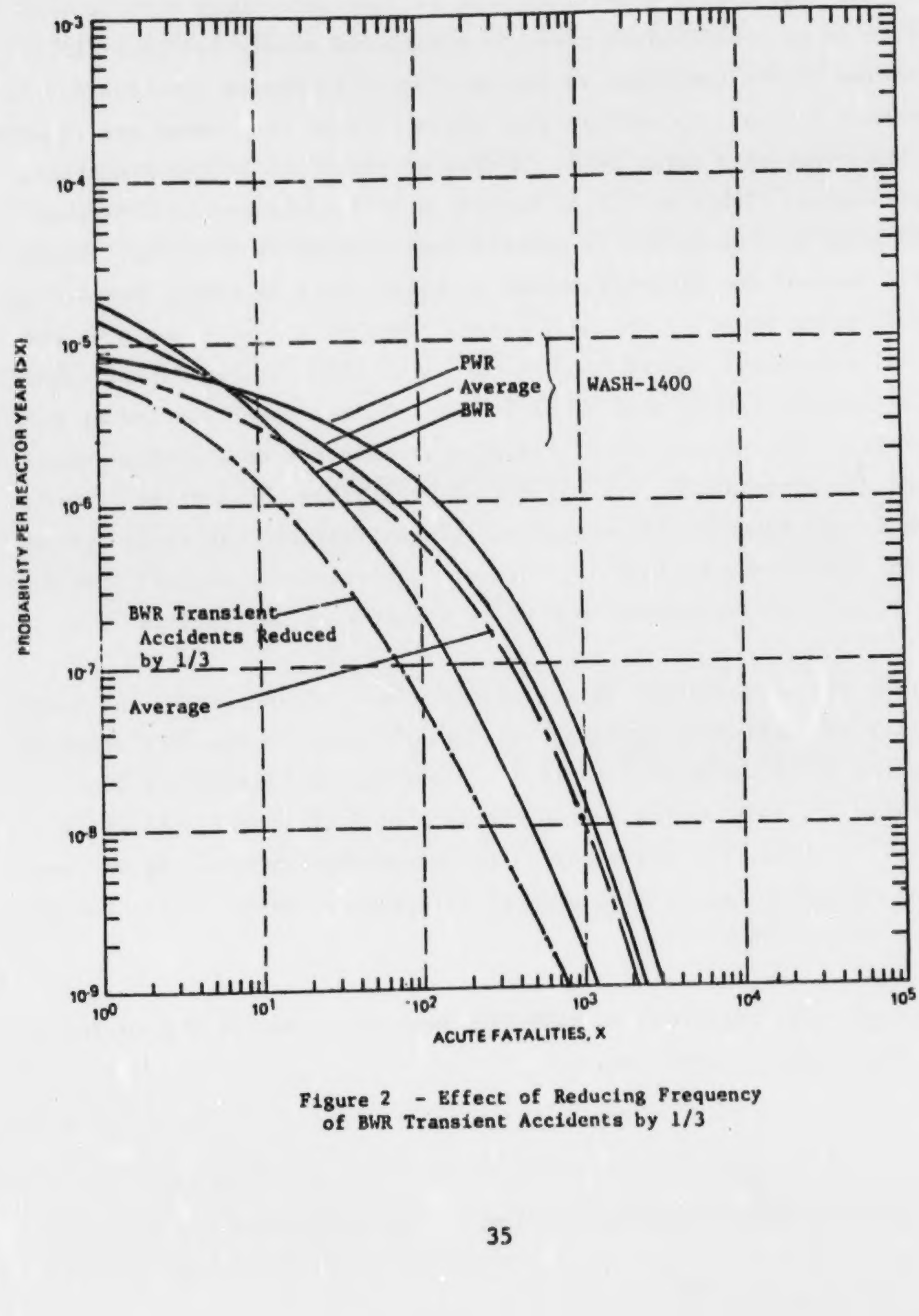


Figure 2 - Effect of Reducing Frequency of BWR Transient Accidents by 1/3

required emergency (diesel) power for providing sufficient low pressure coolant injection (LPCI) capacity during large loss-of-coolant accidents.

The discussion on pages 12, 34, and 35 of Appendix II, Vol. 3, implies that all four diesel generators must be lost coincident with loss of net to cause insufficient power to engineered safety features (ESFs). A review of the Low Pressure Coolant Injection System (Section 6.4.2, Appendix II, Vol. 3) reveals that three of four LPCI pumps are required in the event of a large LOCA. Review of the Peach Bottom electrical distribution (Table 8.5.2b of Reference 8) indicates each LPCI pump (RHR pump in that table) is powered from a separate 4 kV bus. Therefore, loss of two diesel generators, hence two 4 kV buses, results in insufficient power to the LPCI pumps. Loss of off-site net plus two diesel generators should then be the event which causes insufficient LPCI capacity rather than total loss of all a-c power (including four diesels). The discussion of tripping diesels due to starting surges (page 35, Appendix II, Vol. 3) should, therefore, be modified. Section 5.3 of Appendix III states the probability of this event for two diesel generators is 1×10^{-3}. This will increase the overall probability of insufficient power to ESFs by a factor of 10, to 1×10^{-5}.

A study was performed to determine the effect of increasing the probability of sufficient emergency electrical power to the LPCI pumps by a factor of 10, from 10^{-6} to 10^{-5}. According to Section 6.4.2 of Appendix II, Vol. 3, the median estimate of LPCI unavailability is 1.5×10^{-2}. Thus, the contribution to unavailability from the emergency power system is negligible, even if increased a factor of 10 from the value used in WASH-1400.

Omissions - No omissions in WASH-1400 were found during the review of the BWR electric power system.

Conclusions - Correcting the probability of BWR emergency power failure by increasing the WASH-1400 value by a factor of 10 will result in an insignificant increase in calculated risks from BWR accidents.

Miscellaneous Comments - The following comments on WASH-1400 were developed during the course of the review of the BWR electric power system failure analysis:

(1) Appendix II, Vol. 3, Section 6.1 - The common mode failure of all four diesels due to blockage of the cooling water return line may be a significant contributor. It appears that this failure has not been considered in Appendix II.

(2) Appendix II, Vol. 3, Page 43 - It is not clear why Q'(EDG) (failure of emergency diesel generator) is calculated by computing an average between a diesel being unavailable and a running diesel failing. It seems that either failure mode results in the diesel being unavailable and the probabilities of the two modes should be added.

(3) Appendix II, Vol. 3, Section 6.4.2 - It is not clear why the Q_{MED} for total LPCI system unavailability ($1.5\text{x}10^{-2}$) is lower than one of its contributors, $Q_{\text{Test \& Maint.}} = 1.7\text{x}10^{-2}$.

4. PWR Electric Power System Failure

General Review - The failure modes and analysis presented in Appendices II (Vol. 2) and III of WASH-1400 were reviewed. No errors leading to a definitive, significant change in PWR risks due to electric power failure were found. However, a significant number of minor errors, discrepancies, and questionable assumptions were found. In some cases, the effect of the error could not be definitely assessed due to a lack of information. In all such cases, it was judged qualitatively that the effect would not be significant; however, additional effort is required to provide sufficient credibility to such judgments. The errors, discrepancies and questionable assumptions are as follows:

(1) Availability of Power at LOCA - The assumption in Appendix II, Vol. 2, Section 5.1 that all emergency buses are available immediately prior to LOCA is questionable. The basis for this assumption is that the Technical Specifications require reactor shutdown if an emergency bus becomes unavailable. The following should be considered:

   (a) There is a finite probability that Technical Specifications will be violated, either intentionally or unintentionally. A number of AEC-reported "abnormal occurrences" have involved violation of Technical Specifications. Such violations would probably be grouped under the heading of "human reliability," with values of the same order of magnitude as discussed in Section 6.1 of Appendix III (10^{-2} to 10^{-4}). Unavailability of a bus, coupled with violation of a Technical Specification, may be an insignificant contributor to total unavailability, but until quantified, an assessment cannot be made.

   (b) The indicators and annunciators designed to tell the operator he is without a bus have a finite probability of failure. Such a failure is considered for HPIS failure on the fault tree on Figure II-70, Appendix II, Vol. 2, and assigned a point unavailability value of 1.1×10^{-4}. This probability, although minor, does not appear to be considered for the PWR Electric Power System Failure.

(2) Detection of Bus Failures - The assumption (Appendix II, Vol. 2, Section 51) that faults do not exist on a bus at inception of LOCA may be questionable. The methods and frequencies of detecting failures (annunciators, ground detectors, maintenance, etc) will strongly influence the validity of this assumption. Another

consideration should be the operator action required if a ground is detected on the bus. The Technical Specifications do not specify any required action. However, normal ground isolation procedures involve sequentially de-energizing selected buses until the ground is found. There is also a finite probability that undetected faults exist on a bus. Until these considerations are quantified, insufficient data exist to determine the impact that such bus faults would have.

(3) Time Invariant Failure Rate - In the cumulative failure probability discussion on page 30, Appendix II, Vol. 2, the assumption that effective failure rate (λ) remains constant with time appears to be neither conservative nor realistic. In particular, grounds and faults associated with grounds resulting from a harsh LOCA environment would be expected to increase with time. An AEC evaluation(15) of eight LOCA incidents in BWRs showed that grounds occurred during two of the eight incidents. During one of the incidents, grounding of components resulted in loss of capability of control room annunciators to properly indicate the status of both the ECCS and plant radiation instruments. The eight incidents involved primary coolant releases of relatively short duration. Extended exposure to a LOCA environment might be expected to cause long range degradation of electrical insulation and/or components.

(4) Operator Error, Breaker Opening - Appendix II, Vol. 2, Section 5.1 assigns a probability of 10^{-4} to the operator inadvertently opening breaker 15H3 or 15H8 under stress. Such a value does not appear in Table III-13, nor does corresponding discussion appear in the "Human Reliability" section of Appendix III. Furthermore, if the operator is required to take any specific actions during a LOCA which involves operating breakers from the control room, a probability of 10^{-4} for improper action seems to be quite low

compared with the probability of operator errors described in Table III-13.

(5) Battery Unavailability - From the discussion of d-c bus unavailability on pages 26 and 39, Appendix II, Vol. 2, it is not clear what types of faults lead to the unavailability of station batteries (assigned a value of 10^{-3}). Since faults leading to battery unavailability preclude use of the d-c bus, then mere unavailability of the d-c bus will cause insufficient power to ESF. Availability of net becomes insignificant because, even though net power is supplied to the 4160-volt and 480-volt emergency buses, d-c control power is not available to start the ESF loads. Therefore, further discussion and clarification is needed to evaluate the major contributing factors to q (battery) since it can be a significant factor in determining the probability of power to ESF.

Omissions - Several omissions were found in the WASH-1400 analysis of the PWR electric power system. It is possible that some failure modes discussed herein were considered but rejected due to low probability of occurrence or due to insignificant consequences. It is also possible that the failure modes were considered but deleted when the fault trees were simplified. A discussion should be provided in the text if failure modes fall into these categories to give some reasonable assurance of completeness. It does not appear that any of the omissions found would significantly contribute to the overall risk, but further analysis is required to substantiate this conclusion. The following omissions were found:

(1) Operator Error, Opening of Breakers - Appendix II, Vol. 2, Section 5.1 assigns a value of 10^{-4} to the operator inadvertently opening breakers 15H3 or 15H8 under stress. It seems logical to assume that an equal probability exists for him to open other breakers from the control room. These include breakers 15E1,

15C2, 15F1, 15H7, 14H1, 14H10, 14H6, etc, each of which may deenergize its respective emergency bus. Addition of these events to the fault trees of Figure II-23, Sheets 2 and 3, increases the probability of insufficient power to each bus by a factor of two to three.

(2) Maintenance Caused Unavailability - Operator induced maintenance errors and unavailability due to maintenance do not appear on the fault trees. It is possible that no maintenance is performed on the electrical system during operation. However, maintenance errors, such as improperly racking breakers in or miscalibration of relays in control circuits, could contribute to bus unavailability.

(3) Diesel Generator Unavailability - Due to simplification of the fault tree in Figure II-23, Sheet 8, it is impossible to tell if the considerations for diesel generator availability, discussed on pages 35-38, Appendix II, Vol. 2, are included. The importance of the diesel generator warrants inclusion of its subtree as a separate sheet in Figure II-23.

(4) Common Mode Failures - Some common-mode failures appear to have been overlooked or discarded in the analysis. These include the following:

(a) It is possible that an earthquake, equal to or greater than the safe shutdown earthquake, would cause a loss of both onsite and offsite power. Appendix X states that neither the a-c nor the d-c switchgear could be assessed as to seismic design adequacy. On the basis of this statement, it appears that common mode failure due to an earthquake should be evaluated in the fault tree.

(b) A relay failure in the circuitry of breakers 15H8 and 15H3 could result in placing the emergency diesel generator on the 4160v emergency bus while the bus is still energized by the preferred (offsite) source. If the diesel is paralleled out of phase with the offsite source, excessive torque and current can result, tripping both sources. This may be significant since failure of a relay to energize is assigned a value of $1x10^{-4}$/Demand in Table III-1.

(c) Physical location of various ESF load breakers should be evaluated. If a short circuit exists across a load breaker, it may result in drawing excessive current across the breaker and creation of an electric fireball within the switchgear. This fireball, by virtue of its heat and high current, can cause the other breakers in the vicinity (within the same switchgear) to trip open. It is not clear from the discussion that such an evaluation was made.

Trojan Comparison - Comparison of the Surry electric power system with that of Trojan revealed several differences. It appears that the differences would result in a higher value of electric power system reliability for the Trojan reactor than for Surry. The amount of increase is probably small, but an in-depth assessment is required before the difference can be quantified. The differences are as follows:

(1) A major contributor to loss of all electrical power was loss of net, due to challenging the system's transient stability limit. The probability (10^{-3}) that offsite power would be lost as a result of LOCA is based on "generalized information for plants east of the Rockies." The applicability of such a data base to Trojan (or other plants west of the Rockies) may not be valid.

(2) Surry uses three emergency diesels to supply two plants, while Trojan uses two diesels for a single plant. A failure or main-

tenance outage of the "shared" Surry diesel results in two plants losing diesel redundancy, while at Trojan failure or maintenance on a diesel affects only one plant. This results in a slight increase in diesel availability at Trojan.

(3) The emergency loads that each diesel generator is required to supply at Surry comprise a total of 2320 kW, or 85 percent of the diesel rating of 2750 kW. The Trojan emergency loads comprise 3364 kW or 76 percent of the 4416 kW ratings. These numbers indicate that there should be a slightly higher probability of overloading the Surry diesel when it picks up emergency loads than for the Trojan plant. However, other considerations such as overloading which depends on the magnitude of the starting current surges for various components must be considered. A related factor, the diesel generator loading sequence, seems to be spread over a slightly longer time span for Trojan. This should reduce the probability of overloading the diesel if required during an accident.

(4) The d-c system of Trojan is slightly more redundant than Surry. Each of the two battery chargers supplying each battery is powered from a separate motor control center at Trojan, while at Surry the same motor control center supplies both chargers for each battery.

(5) Two variations in the Technical Specifications between Surry and Trojan were found. A primary offsite source can be unavailable for up to 7 days at Surry while only 48 hours is allowed at Trojan before shutting down. Conversely, Trojan is allowed to operate without a power source to an emergency bus for up to 48 hours, while Surry requires shutdown if any emergency bus is deenergized. The first variation causes the Trojan electric power system to be more reliable, while the latter results in less reliability. (It should be noted that the Trojan Technical

Specifications used in this review are those currently included in the Trojan FSAR. They have not been approved for use during reactor operation.)

Conclusions - The following conclusions are derived from the review of the PWR electric power system:

(1) Although numerous inconsistencies, minor errors, and questionable assumptions were found in the WASH-1400 failure analysis of the PWR electric power system failure, none appeared to have the potential for materially changing the overall PWR risk assessment. More detailed analysis is required, however, to quantitatively substantiate this conclusion.

(2) Some minor omissions were found in the WASH-1400 PWR electric power system failure analysis. None appeared to have the potential for materially changing the results of the analysis.

(3) Due to several differences noted between the Surry and Trojan electric power systems, it is not possible to conclude that the results of the Surry analysis applies to the Trojan type reactor. It appears that the electric power system availability for the Trojan class of reactor may be slightly better than for Surry.

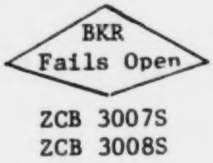
Miscellaneous Comments - The following comments on WASH-1400 were developed during the course of the review of the PWR electric power system:

(1) General - Regulatory Guide 1.93(16) concerning the availability of electric power sources has recently been published. The influence that this Regulatory Guide would have on the availability of power sources should be evaluated. For example, the Regulatory Guide allows operation of a nuclear plant at reduced power levels for specified time periods following loss of an offsite or onsite

power source, even though Technical Specifications might prohibit such operation. Therefore, the initial assumption that offsite power is available at inception of LOCA may not be entirely valid.

(2) Appendix II, Vol. 2, Page 27 - It is not clear in the battery capability discussion whether some minimum battery specific gravity is assumed. The Surry Technical Specifications do not specify a minimum allowable specific gravity. Operating procedures may include such a requirement, but the text discussion should address this subject.

(3) Appendix II, Vol. 2 - The legend provided in Tables II-5, II-6, II-7, and II-8, and the discussion on page 36, Appendix II, Vol. 1 do not correspond to the symbols shown on the fault trees. As an example, the middle fault tree of Figure II-23, Sheet 3, contains a symbol showing:



The prefix "Z" is not shown as a system in Table II-5 while the suffix "S" indicates a short to ground in Table II-8 rather than failure by opening. Consistency with the rest of the study is needed in the use of failure mode symbols.

(4) Appendix II, Vol. 2, Pages 9 & 10 - It is not clear why failures of the emergency power systems were not evaluated for the period between 24 hours and one month after LOCA. It may be an error in the text.

(5) Appendix II, Vol. 2, Pages 46 to 55 - Values of "λ" and "q" were unavailable in many cases except in an abbreviated form on these pages. The Loss of Power Transient did not have a table such as

II-12, to provide the values used in individual fault trees. In many cases, failure values used in the trees were modified versions of data obtained from Table III-1. Modifications were based on "engineering judgment," and although the engineering judgment may be sound, there is no supporting discussion in the text.

5. PWR High Pressure Injection System (HPIS) Failure

General Review - The failure modes and analysis of the PWR high pressure injection system presented in Appendix II, Vol. 2, Section 5.6.4 of WASH-1400 were reviewed. The following discrepancies were found:

(1) Operator Failure to Open Service Water Valves to Lube Oil Coolers - As pointed out on page 313 of Appendix II, Vol. 2, the operator must manually open a valve in the service water system to the standby pump lube oil coolers in the event that a standby pump is required. Otherwise, the pump will fail from overheated lube oil. This operator action is required any time that either standby HPIS pump is needed to protect the core from overheating during a small break LOCA. This can occur under the following circumstances: (1) when failure of the operating pump occurs after a small break in any part of the primary system except the "reactor vessel cavity," and (2) when a small break occurs in the reactor vessel cavity. In the latter case, two pumps are required; therefore, even if the operating pump does not fail, the operator must open at least one of the valves to allow service water flow to the lube oil coolers. The probability that the operator fails to open the service water valves to the two standby pump lube oil heat exchangers is assigned a value of 1×10^{-3} in Appendix II. The value appears to be low based on information presented in Appendix III of WASH-1400. The ability of an operator to correctly respond to a given accident situation is dependent on many factors, as indicated in Appendix III. Dominant among these

factors are (1) the stress level as perceived by the operator when the accident occurs, and (2) the time interval allowed for the action following the onset of the stress condition. The occurrence of a small break is likely to produce a high stress condition for the operator, especially if the break is large enough to create a pressure transient in the system which significantly alters instrument readings in the control room and trips several annunciators. The description of the operator action required to open the valves, the failure of which is assumed to result in failure of the pump, states "Service water valves to lube oil cooler not opened by operator when pump starts" (Item FXVPASWO on page 348 of Appendix II, Vol. 2). The pump is started automatically by the safety injection control system upon receipt of signals of low pressure and low pressurizer level. Thus, for all but very small breaks, the pump will be started quickly. In the event of a small break, it is likely that a high stress condition will be perceived by the operator, and he will be required to act soon after the break if a standby HPIS pump is required. Under these conditions, if the operator is required to act within 60 seconds, his estimated error rate is approximately 1.0 according to Appendix III (page 130) of WASH-1400. However, despite the failure statement for FXVPASWX, the pump will undoubtedly operate for some time before its lube oil becomes hot enough to cause failure. If this time interval is longer than 30 minutes, the error rate in an extreme stress condition is 10^{-1} (page 131). However, a small break probably won't create an "extreme" stress condition. The error rate for a "high" stress condition for action required after several hours is 10^{-2} according to Appendix III (page 131). It would appear then that the proper error rate for the operator during a small break is between 10^{-1} and 10^{-2}. For the purpose of this discussion, a rate of $5x10^{-2}$ will be used.

The operator failure probability to open the service water valve to the lube oil heat exchanger for the second standby pump given that he fails to open the valve to the first standby pump is assumed to be also 10^{-3} in the Appendix II analysis for the HPIS. In Appendix III, page 130, an error rate of approximately 1.0 is given: "If an operator fails to operate correctly one of two closely coupled valves or switches in a procedural step, he also fails to operate the other valve." This description appears to apply to the situation under discussion.

Summarizing, the following table (Table 11) compares the failure (or error) probability given in Appendix II, Vol. 2 for the failure of the operator to open the service water valves to the heat exchanger in the HPIS with what is judged to be a more reasonable assessment.

Table 11

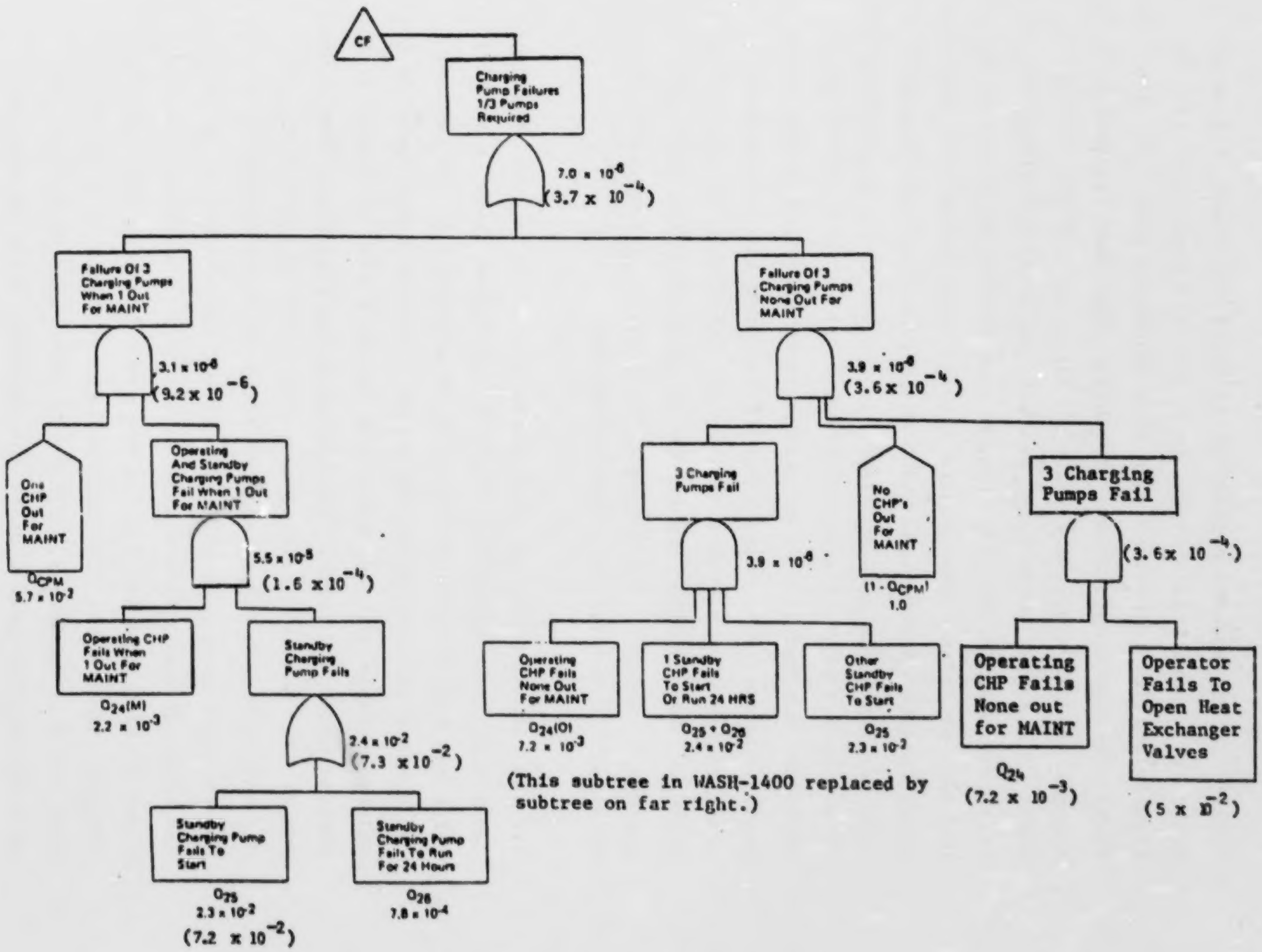
ERROR RATE: SERVICE WATER VALVES TO LUBE OIL COOLER NOT OPENED BY OPERATOR WHEN PUMP STARTS (FXVPASWX)

Event	WASH-1400 Error Rate	As Assessed Herein
1. Operator fails to open first valve	10^{-3}	$5x10^{-2}$
2. Operator fails to open second valve	10^{-3}	1

To determine the significance of the differences in Table II-6A, it is necessary to analyze appropriate fault trees in Appendix II, Vol. 2. The operator error being considered is included in a failure group labeled Q_{25}, a group which includes seven events of the same type as the operator failure being analyzed. The total for failure group Q_{25} (with FXVPASWX = $1x10^{-3}$) is $2.3x10^{-2}$. Using a value of $5x10^{-2}$ for FXVPASWX increases Q_{25} to a value of

7.2×10^{-2}. Q_{25} appears explicitly in two fault trees, Figure II-63, page 371 and Figure II-64, page 372 of Appendix II, Vol. 2; Figure II-63 is reproduced with alterations as Figure 3. The fault tree top event failure probability has been recomputed using a value for Q_{25} of 7.2×10^{-2}. Also, the subtree for the event "3 charging pumps fail" has been replaced to reflect the fact that operator failure to open both service water valves to the lube oil heat exchangers is 5×10^{-2}. This completely dominates other failure combinations which could result in both standby pump failures. As seen in Figure 3, the top event probability is raised by a factor of about 50 (from 7.0×10^{-6} to 3.7×10^{-4}) when revisions are made. (All revised numbers are shown in parentheses.) However, when this change is factored into the overall availability of the system (Figure 4), the change becomes insignificant, from 3.6×10^{-3} (when one of three charging pumps required) to 4.0×10^{-3} (revised numbers shown in parentheses) since other single and double failures dominate.

For the case when two of three pumps are required, the applicable fault tree from Appendix II, Vol. 2, is Figure II-64 (page 372) which has been reproduced and revised as Figure 5. As before, the numbers in parentheses are revised failure rates based on the values in Table 11. Starting from the left side of Figure 5, the subtree under the event, "Failure of 2 Charging Pumps when 1 Out for MAINT," has been revised to reflect the increase in $Q_{25} + Q_{26}$ under "Standby Charging Pump Fails to Start or Run 24 Hrs." On the right hand side of Figure II-6C, the value 2 $(Q_{25} + Q_{26})$ has been increased from 4.8×10^{-2} to 1.5×10^{-1} to reflect the increased value used for Q_{25}. A new event has been added at the extreme right of the figure, "Operator Fails to Open Valves to Lube Oil Heat Exchanger." This has been added to reflect the unit probability used that if the operator fails to open the first valve, he will also fail to open the second, thus disabling both pumps. Since this event is (incorrectly) included under the



CF
Charging Pump Failures 1/3 Pumps Required
7.0×10^{-6}
(3.7×10^{-4})
Failure Of 3 Charging Pumps When 1 Out For MAINT
3.1×10^{-6}
(9.2×10^{-6})
One CHP Out For MAINT
Q_{CPM}
5.7×10^{-2}
Operating And Standby Charging Pumps Fail When 1 Out For MAINT
5.5×10^{-5}
(1.6×10^{-4})
Operating CHP Fails When 1 Out For MAINT
$Q_{24}(M)$
2.2×10^{-3}
Standby Charging Pump Fails
2.4×10^{-2}
(7.3×10^{-2})
Standby Charging Pump Fails To Start
Q_{25}
2.3×10^{-2}
(7.2×10^{-2})
Standby Charging Pump Fails To Run For 24 Hours
Q_{26}
7.8×10^{-4}
Failure Of 3 Charging Pumps None Out For MAINT
3.9×10^{-6}
(3.6×10^{-4})
3 Charging Pumps Fail
3.9×10^{-6}
No CHP's Out For MAINT
$(1 - Q_{CPM})$
1.0
Operating CHP Fails None Out For MAINT
$Q_{24}(O)$
7.2×10^{-3}
1 Standby CHP Fails To Start Or Run 24 HRS
$Q_{25} + Q_{26}$
2.4×10^{-2}
Other Standby CHP Fails To Start
Q_{25}
2.3×10^{-2}
(This subtree in WASH-1400 replaced by subtree on far right.)
3 Charging Pumps Fail
(3.6×10^{-4})
Operating CHP Fails None out for MAINT
Q_{24}
(7.2×10^{-3})
Operator Fails To Open Heat Exchanger Valves
(5×10^{-2})

Figure 3 - Revised Quantitative Pictorial Summary for One of Three HPIS Pumps with Point Failure Estimates

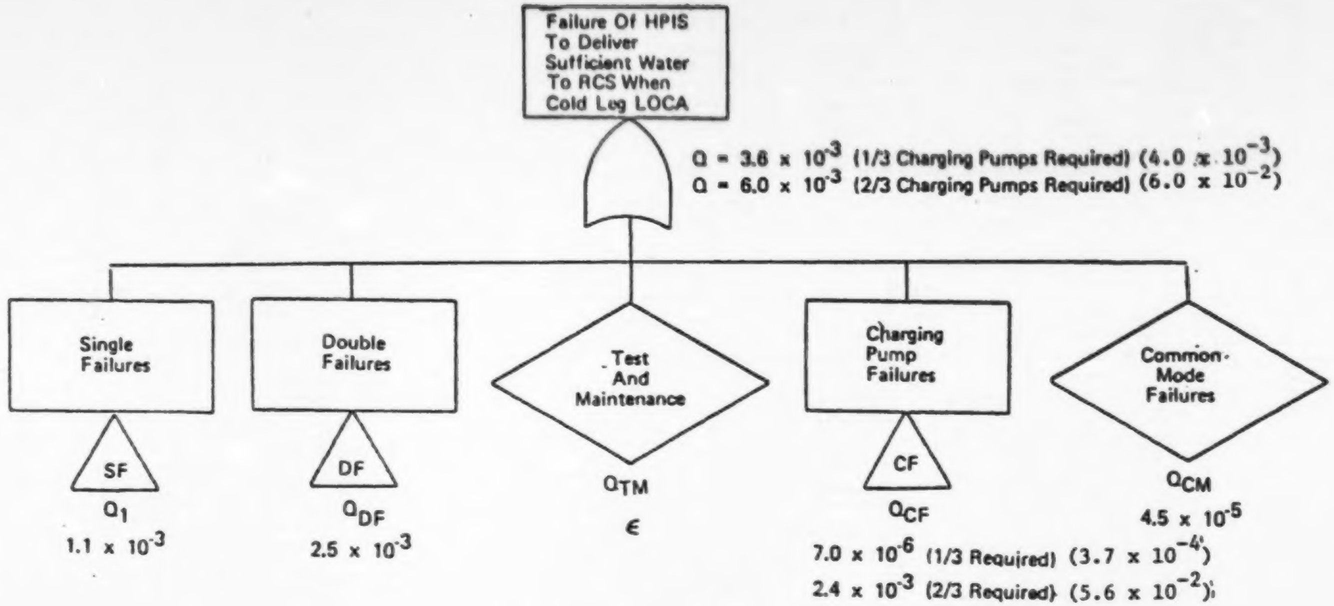

Failure Of HPIS To Deliver Sufficient Water To RCS When Cold Leg LOCA
Q = 3.6 x 10^{-3} (1/3 Charging Pumps Required) (4.0 x 10^{-3})
Q = 6.0 x 10^{-3} (2/3 Charging Pumps Required) (6.0 x 10^{-2})
Single Failures
SF
Q_1
1.1 x 10^{-3}
Double Failures
DF
Q_{DF}
2.5 x 10^{-3}
Test And Maintenance
Q_{TM}
ϵ
Charging Pump Failures
CF
Q_{CF}
7.0 x 10^{-6} (1/3 Required) (3.7 x 10^{-4})
2.4 x 10^{-3} (2/3 Required) (5.6 x 10^{-2})
Common-Mode Failures
Q_{CM}
4.5 x 10^{-5}

Figure 4 - Quantitative Pictorial Summary of HPIS Failure With Point Estimates Showing Effect of Q_{cf} Increase

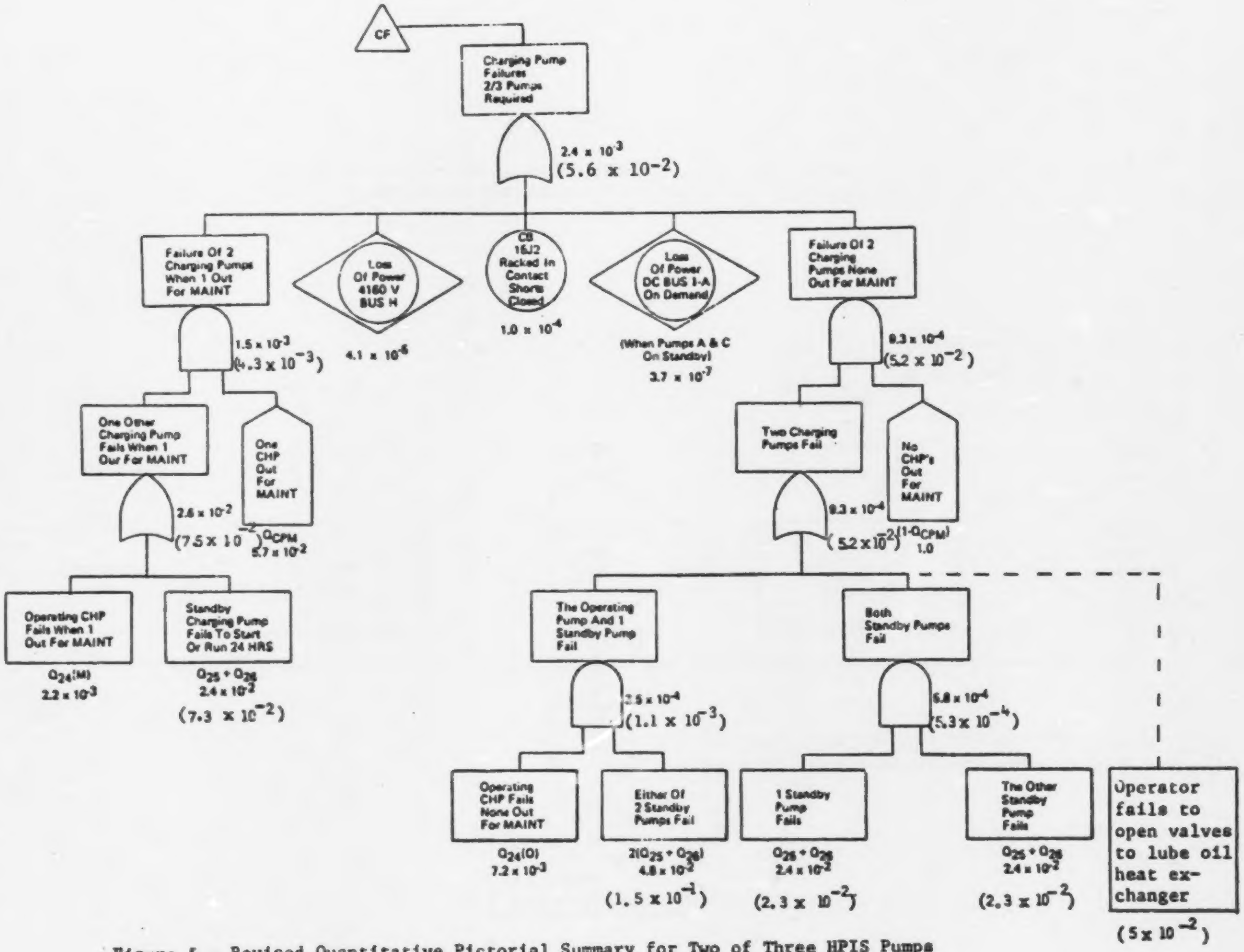

CF
Charging Pump Failures 2/3 Pumps Required
2.4 x 10-3
(5.6 x 10-2)
Failure Of 2 Charging Pumps When 1 Out For MAINT
1.5 x 10-3
(4.3 x 10-3)
Loss Of Power 4160 V BUS H
4.1 x 10-6
CB 16J2 Racked In Contact Shorts Closed
1.0 x 10-4
Loss Of Power DC BUS 1-A On Demand
(When Pumps A & C On Standby)
3.7 x 10-7
Failure Of 2 Charging Pumps None Out For MAINT
9.3 x 10-4
(5.2 x 10-2)
One Other Charging Pump Fails When 1 Our For MAINT
2.6 x 10-2
(7.5 x 10-2)
One CHP Out For MAINT
QCPM
5.7 x 10-2
Two Charging Pumps Fail
9.3 x 10-4
(5.2 x 10-2)
No CHP's Out For MAINT
(1-QCPM)
1.0
Operating CHP Fails When 1 Out For MAINT
Q24(M)
2.2 x 10-3
Standby Charging Pump Fails To Start Or Run 24 HRS
Q25 + Q26
2.4 x 10-2
(7.3 x 10-2)
The Operating Pump And 1 Standby Pump Fail
2.5 x 10-4
(1.1 x 10-3)
Both Standby Pumps Fail
5.8 x 10-4
(5.3 x 10-4)
Operating CHP Fails None Out For MAINT
Q24(O)
7.2 x 10-3
Either Of 2 Standby Pumps Fail
2(Q25 + Q26)
4.8 x 10-2
(1.5 x 10-1)
1 Standby Pump Fails
Q25 + Q26
2.4 x 10-2
(2.3 x 10-2)
The Other Standby Pump Fails
Q25 + Q26
2.4 x 10-2
(2.3 x 10-2)
Operator fails to open valves to lube oil heat exchanger
(5 x 10-2)

Figure 5 - Revised Quantitative Pictorial Summary for Two of Three HPIS Pumps With Point Failure Estimates

events "1 Standby Pump Fails" ($Q_{25} + Q_{26}$) as well as "The Other Standby Pump Fails" ($Q_{25} + Q_{26}$), the operator error rate contribution to these values ($1x10^{-3}$) must be subtracted out. This yields the new values as shown in the parentheses. Recomputing the values "up" the fault tree produces a probability for "Charging Pump Failure 2/3 Pumps Required" of $5.6x10^{-2}$ versus a value of $2.4x10^{-3}$ calculated in WASH-1400. This new value has been translated to Figure 6 (Figure II-60 of Appendix II, Vol. 2) and ultimately a failure (Q) of $6.0x10^{-2}$ is computed for "Failure of HPIS to Deliver Sufficient Water to RCS when Cold Leg LOCA" (2/3 Charging Pumps Required). This value is a factor of 10 higher than the WASH-1400 value. The significance of these changes on the ultimate risks to the public are difficult to determine since the distinction between a small break accident which requires one of three charging pumps and one which requires two of three does not appear to have been separately considered in computing the risks. The discussion of small break accidents in Appendix I, pages 123-132 does not acknowledge the "reactor vessel cavity" break as one which requires two of three HPIS pumps. In Appendix V, Table V-7, the failure probability of emergency coolant injection for PWR small breaks (denoted by the letter D) appears to be $1x10^{-2}$. It is not clear how this number was computed, nor whether it included considerations of 2/3 charging pump failures. In any event, it does not appear that the changes discussed here will have a significant influence on the risks. However, the corrections should be made and new risks computed.

Omissions - This review discusses omissions found in the WASH-1400 analysis of the HPIS.

(1) Low Pressure Injection System (LPIS) Check Valve Failure - Connected to the three HPIS injection lines near the injection point to the reactor primary system are three (one in each line) lines from the LPIS. These lines contain check valves near the

connection point with the HPIS lines to prevent, in case of a small break accident, the HPIS from injecting high pressure water into the LPIS. Figure 6 (a portion, simplified and enlarged, of Figure II-65 from Appendix II, Vol. 2) shows the piping and valve arrangement at the intersection of the HPIS and LPIS. It appears that if any of the three LPIS check valves are failed open either before, at the initiation of, or during HPIS injection, some HPIS flow will be diverted into the LPIS. Since the LPIS is designed only for low pressure operation, it will probably rupture. Because the HPIS flow to the three cold legs is connected by a common header, and since the LPIS lines are large (6") relative to the HPIS injection line (3"), a significant portion of the HPIS flow could be diverted in the event of a LPIS check valve failure. It thus appears that a single failure of either of the three LPIS check valves (CV120, CV220, CV320 on Figure 6) will fail the HPIS. The probability of a check valve failure is given as $1.3\text{x}10^{-3}$ on page 339 (dominant fault FCV02200). Assuming that the failure of the check valves are independent, the probability of any one of the three failing is $3\text{x}1.3\text{x}10^{-3}$ or approximately $4\text{x}10^{-3}$. The effect of this single failure mode on the overall system failure mode is shown on Figure 7, a revision of Figure II-60 in Appendix II, Vol. 2. As can be seen, the effect is not very significant, increasing failure probability from $3.6\text{x}10^{-3}$ to $7.6\text{x}10^{-3}$ for the case when one of three charging pumps are required, and from $6.0\text{x}0^{-3}$ to $9.9\text{x}10^{-3}$ when two of three are required.

(2) Additional Double Failures - Three HPIS double failures which may have some significance appear to be ignored in Appendix II, Vol.2. These are:

   (a) Failure of both valves in the drain line of the Volume Control Tank to close after a small break accident (designated valves 1115C and 1115E in Appendix II).

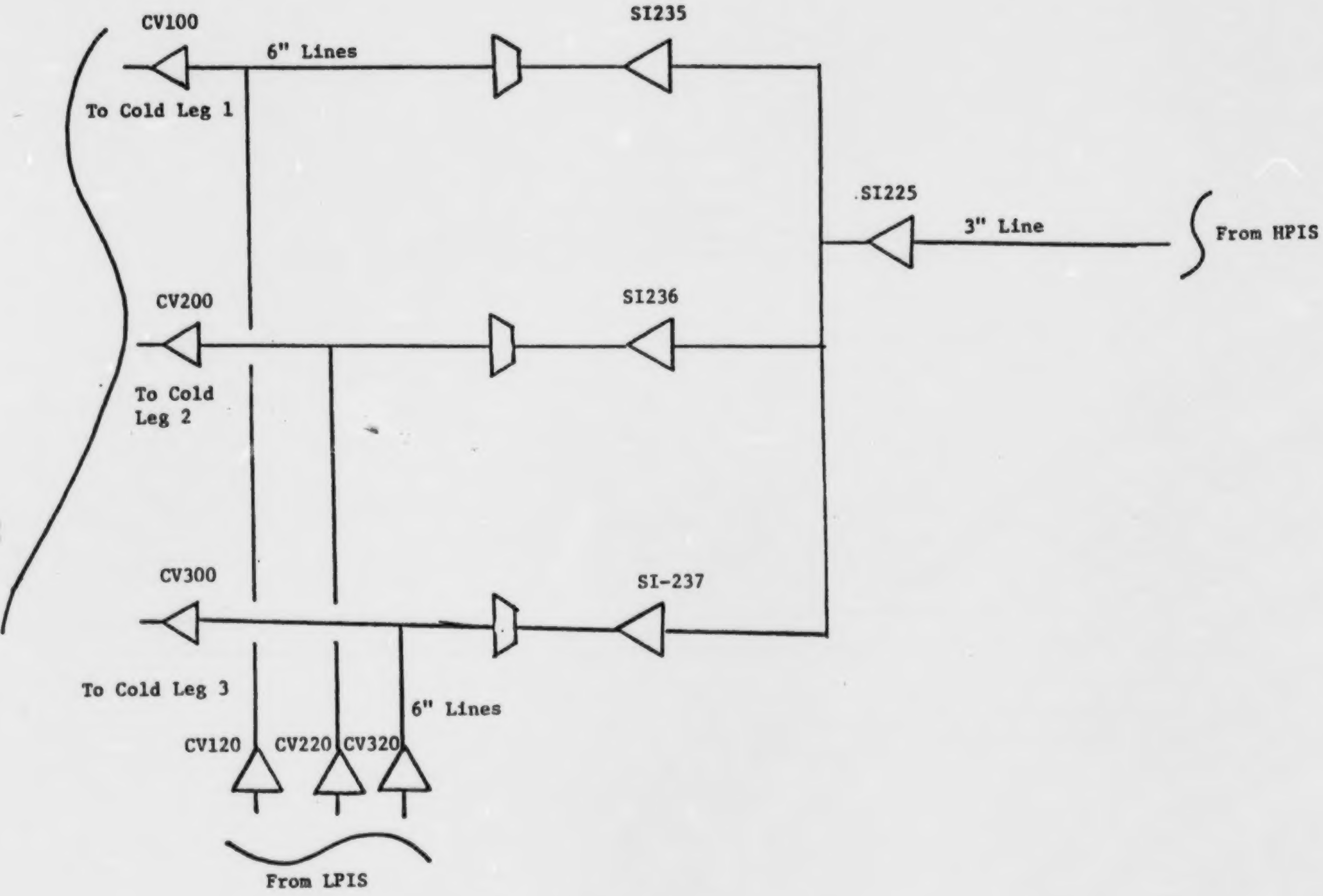

CV100
6" Lines
SI235
To Cold Leg 1
.SI225
3" Line
From HPIS
CV200
SI236
To Cold
Leg 2
CV300
SI-237
To Cold Leg 3
6" Lines
CV120
CV220
CV320
From LPIS

Figure 6 - HPIS - LPIS Piping Intersection Diagram

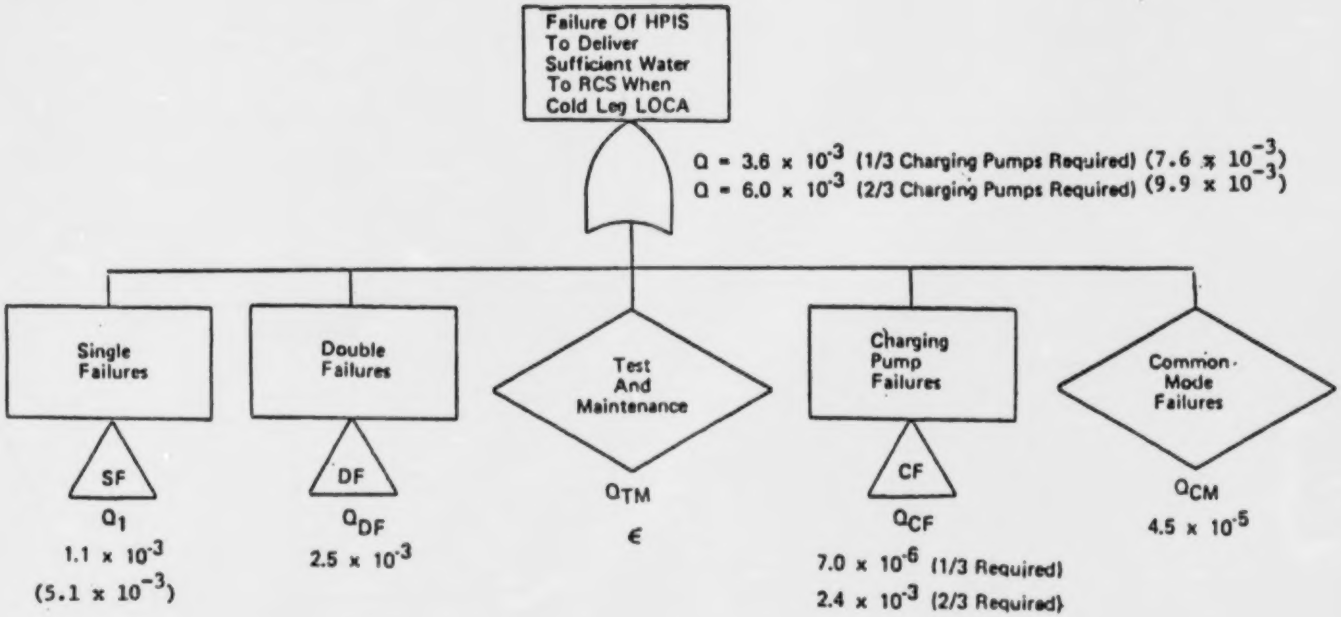

Failure Of HPIS To Deliver Sufficient Water To RCS When Cold Leg LOCA
Q = 3.6 x 10-3 (1/3 Charging Pumps Required) (7.6 x 10-3)
Q = 6.0 x 10-3 (2/3 Charging Pumps Required) (9.9 x 10-3)
Single Failures
SF
Q1
1.1 x 10-3
(5.1 x 10-3)
Double Failures
DF
QDF
2.5 x 10-3
Test And Maintenance
QTM
∈
Charging Pump Failures
CF
QCF
7.0 x 10-6 (1/3 Required)
2.4 x 10-3 (2/3 Required)
Common Mode Failures
QCM
4.5 x 10-5

Figure 7 - Quantitative Pictorial Summary of HPIS Failure With Point Estimates Showing Effect of Q_1 (Single Failure) Increase

(b) Failure of both valves in the normal charging line to close after a small break accident (designated valves MOV 1289A and 1289B in Appendix II).

(c) Failure of both valves in the boric acid recirculation system to close after a small break accident (designated valves 1884A and 1884B in Appendix II).

The closing of the above three sets of valves is part of the automatic sequence to align the HPIS into the injection mode following a small break LOCA. Presumably, the failure of any set of the valves fails the operation of the HPIS, although it is possible that such failures will result in only partial degradation of the system. However, partial degradation of a safety system is normally assigned failure in WASH-1400. Therefore, these double valve failures should be considered in the double failure category.

Figure 8 shows all HPIS double failure contributions considered in Appendix II plus the three sets of double valve failures described previously. The numbers in parentheses are the revised values considering these failures, and assuming a failure probability of 2.0×10^{-2} per valve which seems to be close to the average value assumed for valves failing to open. As can be seen, the increase in double failure probability (from 2.5×10^{-3} to 3.7×10^{-3}) is not significant.

Trojan Comparison - This review consists of a comparison between the Surry reactor system, used in WASH-1400 as a basis to compute risks for all PWRs, and the Trojan reactor system(2), representative of Westinghouse 4-loop designs.

According to the Surry Final Safety Analysis Report(3), protection against small breaks up to 6 inches in diameter is afforded by a high

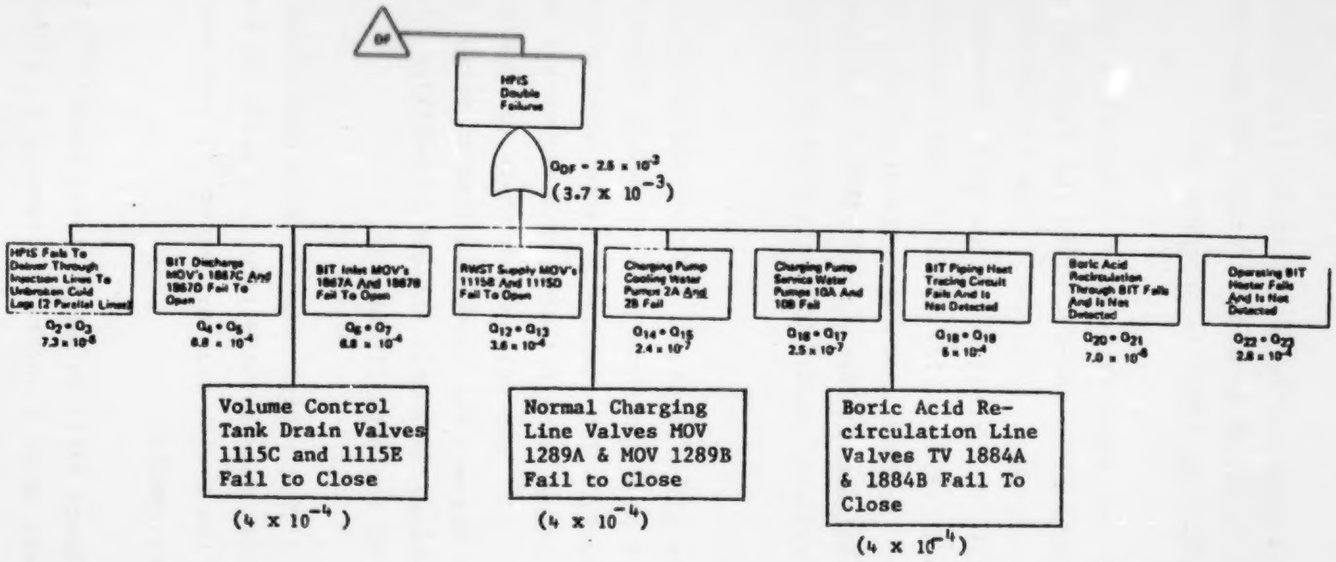

DF
HPIS Double Failure
$Q_{DF} = 2.8 \times 10^{-3}$
(3.7×10^{-3})
HPIS Fails To Deliver Through Injection Lines To Unbroken Cold Legs (2 Parallel Lines)
$Q_2 \cdot Q_3$
7.3×10^{-6}
BIT Discharge MOV's 1867C And 1867D Fail To Open
$Q_4 \cdot Q_5$
6.8×10^{-4}
BIT Inlet MOV's 1867A And 1867B Fail To Open
$Q_6 \cdot Q_7$
6.8×10^{-4}
RWST Supply MOV's 1115B And 1115D Fail To Open
$Q_{12} \cdot Q_{13}$
3.6×10^{-4}
Charging Pump Cooling Water Pumps 2A And 2B Fail
$Q_{14} \cdot Q_{15}$
2.4×10^{-7}
Charging Pump Service Water Pumps 10A And 10B Fail
$Q_{16} \cdot Q_{17}$
2.5×10^{-7}
BIT Piping Heat Tracing Circuit Fails And Is Not Detected
$Q_{18} \cdot Q_{19}$
6×10^{-4}
Boric Acid Recirculation Through BIT Fails And Is Not Detected
$Q_{20} \cdot Q_{21}$
7.0×10^{-6}
Operating BIT Heater Fails And Is Not Detected
$Q_{22} \cdot Q_{23}$
2.8×10^{-4}
Volume Control Tank Drain Valves 1115C and 1115E Fail to Close
(4×10^{-4})
Normal Charging Line Valves MOV 1289A & MOV 1289B Fail to Close
(4×10^{-4})
Boric Acid Recirculation Line Valves TV 1884A & 1884B Fail To Close
(4×10^{-4})

Figure 8 - HPIS Modified Double Failure Contribution Summary

pressure injection flow rate of 150 gpm, equivalent to the design flow rate of one (out of a total of three) charging pump. The Trojan FSAR indicates that a flow rate of 575 gpm (minimum engineered safety equipment - one charging and one safety injection pump) affords protection for the core for breaks up to 6 inches in diameter. Since the Trojan charging pumps (a total of two) are rated at 150 gpm, the total charging system capacity (300 gpm) is less than that assumed to protect the system for all small breaks up to 6 inches. To provide for such protection, the Trojan design includes a safety injection system (SIS), considered part of the HPIS. The SIS consists of two pumps, with associated piping and hardware, each rated at 425 gpm. It thus requires the capacity of at least one charging pump plus one SIS pump to protect the Trojan core for breaks up to 6 inches. Table 12 summarizes the differences in requirements as well as design for the Surry and Trojan HPIS.

It should be noted that different break sizes (up to 6 inches) will require different injection flow rates to protect the core. In the Trojan case, the 575 gpm requirement is based on a conservative analysis of the most demanding (in terms of injection requirements) break size. Not all break sizes up to 6 inches will require a 575 gpm injection flow rate. (The same can be said of the 150 gpm flow rate used for Surry.) However, in the absence of information relative to break size probabilities in the up-to-6-inch size range, as well as incomplete information relative to flow rate requirements as a function of break size, a more detailed assessment for Trojan was not possible, and an analysis similar to that done by WASH-1400 for Surry was used instead.

Table 12 - HPIS COMPARISON
SURRY vs TROJAN

	Surry	Trojan
Minimum flow (gpm)	150	575
System(s)	Charging Only	Charging + Safety Injection
No. of Pumps	3	2 (Charging) 2 (Safety Injection)
Pump Flow Rate (gpm)	150	150 (Charging) 425 (Safety Injection)

The basic design of the HPIS for the two plants is quite similar, and excluding pump failures, comparison of the failure modes for the hardware in the two systems indicates no significant differences exist. However, since the Trojan system requires at least two pumps out of a total of four from two independent sets of two pumps, and Surry requires only one from a single set of three pumps, the Surry HPIS pump failure analysis does not apply to Trojan. The significance of the difference is difficult to assess without undertaking a detailed analysis of the Trojan system. However, an indication may be obtained by considering Table 13 which lists all the pump operating-failure combinations for the two plants. Six out of sixteen of the Trojan combinations result in HPIS failure, while only one of eight Surry combinations fail the HPIS. This may indicate that the Trojan HPIS failure probability is higher than Surry. However, until a complete analysis of the Trojan (or any Westinghouse 4-loop) system is accomplished, including appropriate consideration of pump motor failure rates, common mode failures, etc, no specific conclusions can be supported. In view of the fact that small break accident sequence probabilities are significant contributors to some of the release categories (particularly Categories 3, 4 and 7, this analysis has added importance.

Conclusions - Based on a review of the HPIS failure analysis contained in Appendix II, Vol. 2 of WASH-1400, the following conclusions are derived:

Table 13 - HPIS PUMP SUCCESS-FAILURE COMBINATIONS FOR TROJAN & SURRY

	TROJAN						SURRY				
	Charging Pumps		SIS Pumps		HPIS System		Charging Pumps			HPIS System	
NO.	A	B	A	B	Succeeds	Fails	A	B	C	Succeeds	Fails
1.	runs	runs	runs	runs	X		runs	runs	runs	X	
2.	runs	runs	runs	fails	X		runs	runs	fails	X	
3.	runs	runs	fails	runs	X		runs	fails	runs	X	
4.	runs	runs	fails	fails	X		fails	runs	runs	X	
5.	runs	fails	runs	runs	X		fails	fails	runs	X	
6.	fails	runs	runs	runs	X		fails	runs	fails	X	
7.	runs	fails	runs	fails	X		runs	fails	fails	X	
8.	runs	fails	fails	runs	X		fails	fails	fails		X
9.	fails	runs	runs	fails	X						
10.	fails	runs	fails	runs	X						
11.	fails	fails	fails	fails		X					
12.	fails	fails	fails	runs		X					
13.	fails	fails	runs	fails		X					
14.	fails	fails	runs	runs		X					
15.	runs	fails	fails	fails		X					
16.	fails	runs	fails	fails		X					

(1) There do not appear to be any significant errors in the analysis presented.

(2) There do not appear to be any significant omissions in the analysis presented.

(3) The application of the results of the analysis to Westinghouse reactors of 4-loop design has not been justified.

(4) The minor errors and omissions found need to be corrected and the results carried through to the final risk assessments in the report in order that confidence can be obtained in the results and to assure that such errors are not significant.

Miscellaneous Comments - In reviewing Section 5.6.4 of Appendix II, Vol. 2, the following comments were developed:

(1) Errors

Location	Reads Now	Should Read
Page 326	(see Figure 8)	(see Figure II-58)
Page 328	SW 11130	SW 130
Page 339	CV 236	SI 236
Page 339	866 E	1866 E
Page 373, Figure II-65, Note 1 (A)	LCV 1115 B or LCV 1115 D	MOV 1115 B or MOV 1115D

(2) Page 308, bottom of page - Note 1 indicates, from its placement in the text, that Appendix I discusses the small break located in the reactor vessel cavity. No such discussion can be found in Appendix I.

(3) Page 323, Item 1 - The basis for this conclusion needs to be included in the report.

(4) Page 327, Section 4.5 - The list of significant single failures in this section does not correspond to the single failures quantified on pages 334 and 335. For example, RWST drain plugged and RWST vent plugged (item 13, page 329) do not appear to be considered on pages 334 and 335.

6. PWR Small Break Loss of Coolant Accident Analysis

This review involved the investigation of analytical methods used by PWR vendors to calculate the consequences of a small break LOCA. Since WASH-1400 contains a small break LOCA description, and even more detailed descriptions are readily available(17,18,19), only a very brief discussion, necessary for understanding the subsequent discussion of small break LOCA analysis techniques, will be provided here. A small break is currently defined as a rupture in the PWR primary system which produces an area of 0.5 ft^2 (9-inch diameter circular hole) or less. (This definition does not correspond to that used in WASH-1400; a hole 6 inches in diameter or less, but the difference is not significant.)

Following a small break, the primary system begins to depressurize. When the pressurizer pressure falls below a preset value, the reactor trips (scrams). Other signals cause the HPIS pumps to be started. When the primary system pressure falls below the maximum HPIS injection pressure (around 1800 psi), check valves open and cold water is injected into the primary system. When the system pressure falls to the accumulator injection pressure (about 650 psi), water from the accumulator is injected. At lower pressures, additional emergency cooling water is injected from low pressure injection systems.

For a small break LOCA, the critical time from the standpoint of restoring and maintaining adequate core cooling occurs during the first 1000 seconds or so. During this time period, a complex dynamic interaction is taking place in the core involving energy production, energy redistribution and energy removal. Energy is being produced by the continued fissioning process which is significant until the core is shut down by insertion of control rods or by steam voids which eventually form in the core. Decay heat also contributes to the energy production in the core. Significant energy redistribution occurs in the core when the heat transfer from the fuel pin cladding to the primary coolant is reduced. Energy then accumulates in the cladding and a rise in cladding temperature occurs. One of the most important parameters during this process of energy removal and redistribution is the heat transfer effectiveness from the clad to the coolant. This heat transfer process is governed by coolant properties of pressure, temperature, flow and quality. It is very difficult under the transient conditions existing during the small break LOCA to calculate accurately all of these quantities.

Once the fluid properties have been calculated, heat transfer coefficients must be derived in order to calculate the amount of heat lost from the core cladding. Since there is at present no rigorous, direct way to calculate heat transfer coefficients from basic physical principles, correlations which are primarily based on steady-state experimental results must be used.

The procedure for calculating the core temperature distribution following a small break LOCA is very uncertain. Dynamic fluid property calculations in the core region are very difficult; the application of fluid-condition-dependent heat transfer correlations based on steady-state conditions to the transient fluid conditions calculated to exist in the core is somewhat uncertain, and the correlations themselves are discontinuous and occasionally inaccurate when compared to steady-stat experimental heat transfer data other than that from which they were derived. These uncertainties are allegedly accounted for by conservatism in the calculations.

It should be noted that very little emphasis has been placed on the development of small break LOCA analytical techniques in recent years.

The primary emphasis has been on achieving an adequate analysis for describing the large break LOCA, since it was considered to be the major risk to the health and safety of the general public. WASH-1400, as noted in Section III, concludes that the small break LOCA is a major risk.

Each of the three PWR vendors (Westinghouse, Babcock & Wilcox, Combustion Engineering) has developed an analytical method to calculate the reactor system response following a small break LOCA. Table 14 lists the computer codes used by each vendor for the calculation. One of the basic features of each computational technique is the nodalization scheme employed, as shown in the third column of Table 14. This column describes the number and general location of nodes, or volumetric regions, selected by each vendor for his particular calculation. When the calculation is performed, the thermohydraulic processes are computed separately within each node. The fluid properties, heat transfer from surfaces, fluid flow, etc, are thus identical throughout each node, and the calculated quantities for each node are actually average quantities calculated to exist in the node. As can be seen in Table 14, recent versions of each of the codes contain about the same number of nodes.

A detailed description of each code is not available because many of the basic features are proprietary. Some features of the codes are explained in References 17 through 20, and since these references are readily available, the information contained therein will not be included in this discussion.

One way to assess the capability of the codes in a relative sense is to compare results obtained by the various codes for similar calculations. Since the reactors sold by all three vendors are similar in power level, operating conditions, fluid volume, configuration, etc, it is expected that these calculations would yield generally similar results.

Table 14 - SMALL BREAK LOCA COMPUTER CODES USED BY PWR VENDORS

VENDOR	CODE NAME(S)	NODALIZATION SUMMARY	REMARKS
Westinghouse	SLAP	3 nodes: 1 in steam generator, 1 in pressurizer, 1 in balance of primary system	The WFLASH code was introduced by Westinghouse in May 1974; previously, small break analyses were performed by the SLAP code.
	WFLASH	16 nodes in primary system, 2 in secondary, 1 in pressurizer	
Combustion Engineering	CEFLASH-4	Greater than 15 nodes for primary system*	WALSOB used for hot leg breaks and very small cold leg breaks; CEFLASH-4 used for all other small breaks.
	WALSOB	1 node for primary system	
Babcock & Wilcox	CRAFT, FOAM	14 nodes for primary system,1 node for containment, 1 node for secondary (CRAFT)	The FOAM code computes liquid level in the core region based on input obtained from a CRAFT calculation.

*Actual number of nodes proprietary

As discussed previously, one of the critical parameters in the small break LOCA, from the standpoint of assessing accident risks, is the rate of heat transfer from the core. This rate is strongly dependent on the fluid conditions calculated to exist in the core region. It is important, therefore, to be able to accurately calculate fluid conditions in the core, particularly the pressure history and the core fluid level.

(1) Core Uncovery Time - Figure 9 presents the calculated core uncovery time (i.e., the time following the initiation of the break when the steam-water interface which develops in the vessel drops below the level of the top of the core) for each of the vendors' PWRs. For the three cases, the break location (pump discharge) is the same, the initial primary fluid conditions are very nearly the same, the core locations are similar and the upper plenum volumes are very nearly the same (all 2100 ft^3 ± 120 ft^3). Thus, it would be expected that, except for minor differences due to small design variations, the core uncovery times for the three plants should be comparable. As Figure 9 indicates, the core uncovery times vary substantially from about 80 seconds to 520 seconds for 0.1 ft^2 breaks. For larger breaks, the agreement is somewhat better, but never closer than about a factor of two for the two larger break sizes for which comparative data is available. Figures 10 and 11 show comparisons of the two-phase core fluid level during the transient. Figure 10 is a comparison between two vendors for 0.2 ft^2 pump discharge breaks. Although the general shape of the curves is somewhat similar, they are displaced in time some 150 seconds, a substantial difference. Figure 11 is a similar comparison for 0.87 ft^2 pump discharge break. Again, the shapes are vaguely similar, but displaced in time some 600 seconds, a very large difference.

(2) Pressure History - Figure 12 compares the decompression pressure history as calculated by the three vendors for various break

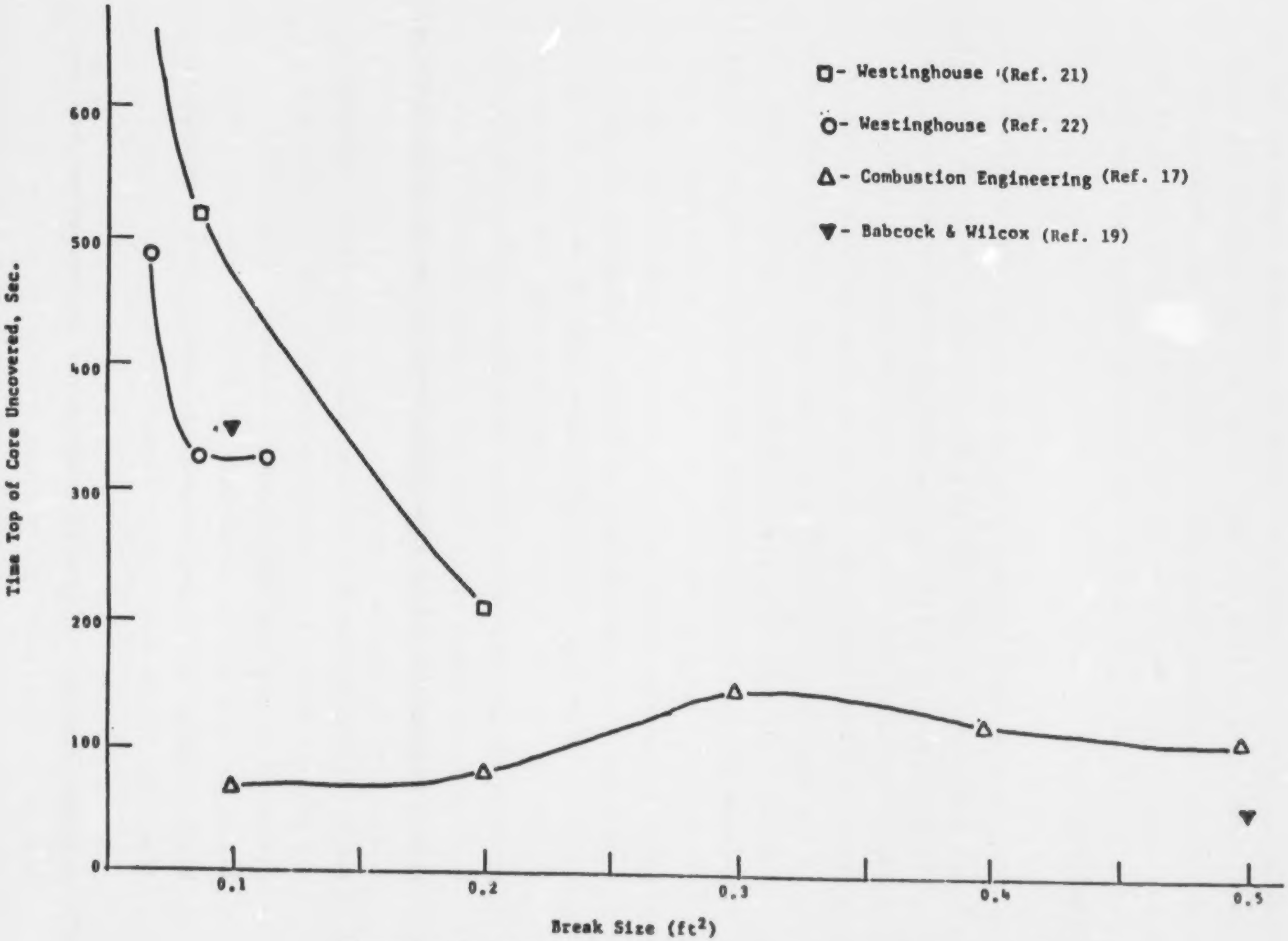


Figure 9 - Time Top of Core Uncovered vs Break Size - Pump Discharge Break

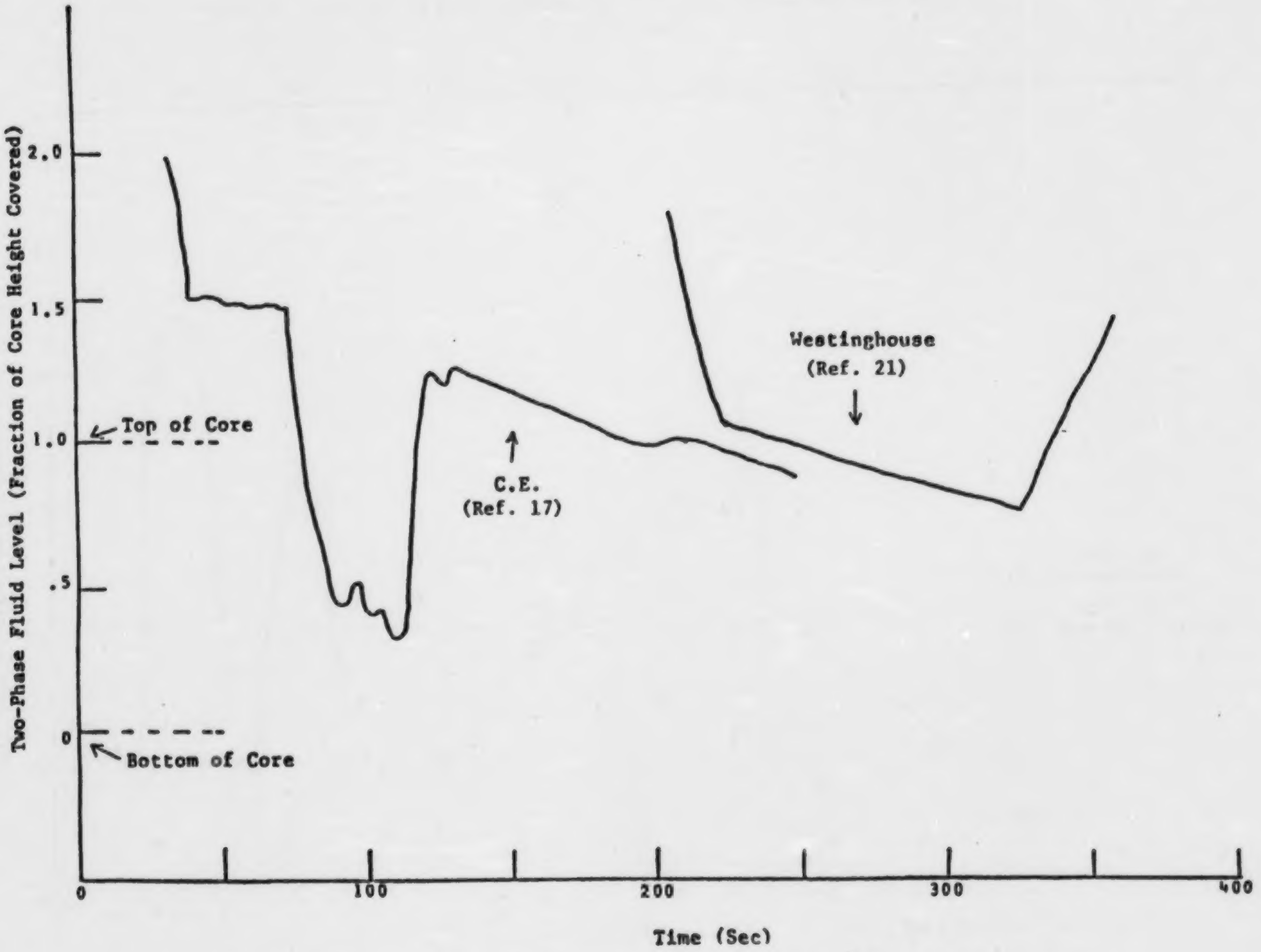


Figure 10 - Two-Phase Fluid Level vs Time for 0.2 Ft^2 Pump Discharge Break

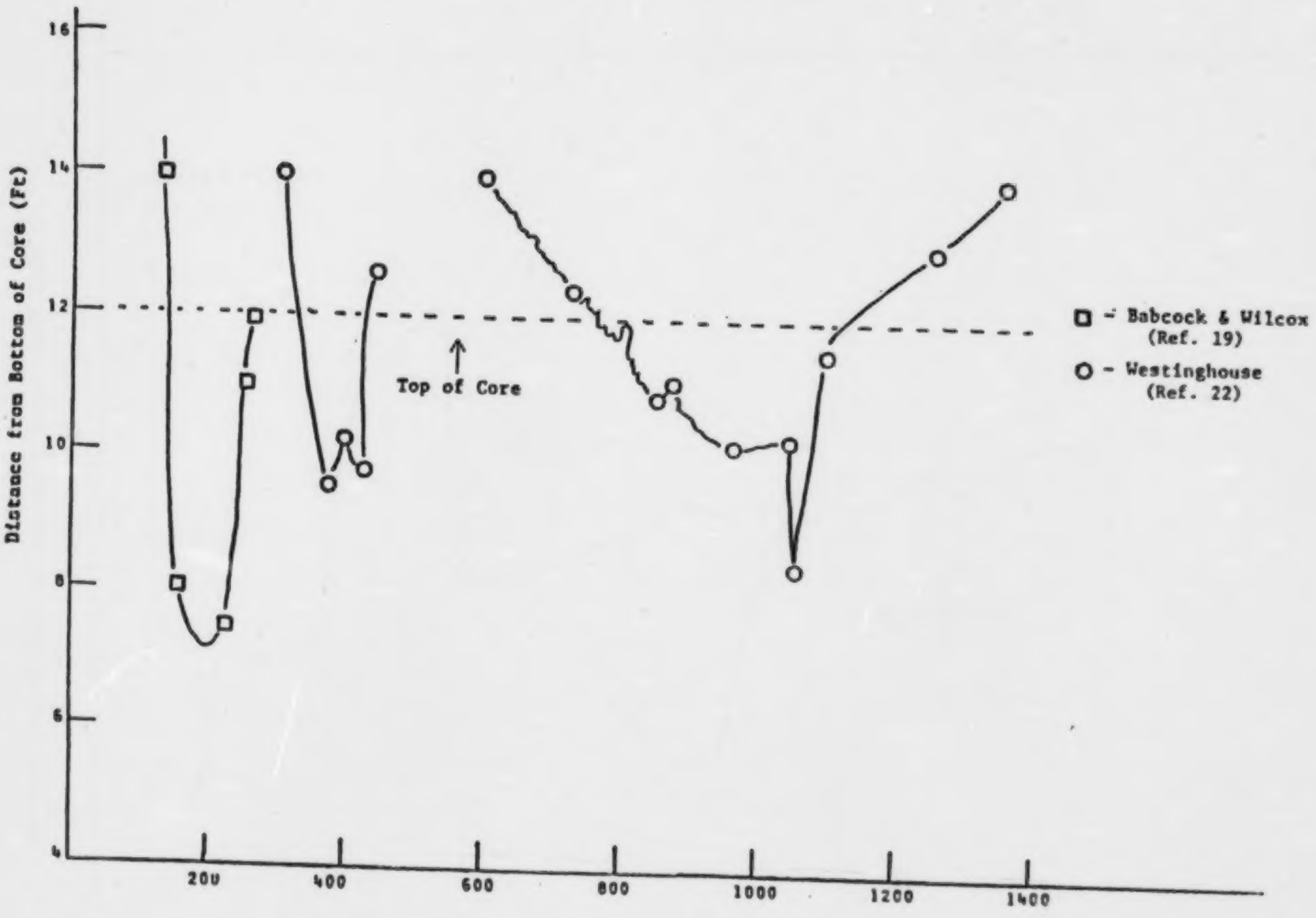


Figure 11 - Core Liquid Level vs Time for 0.087 Ft^2 Pump Discharge Break

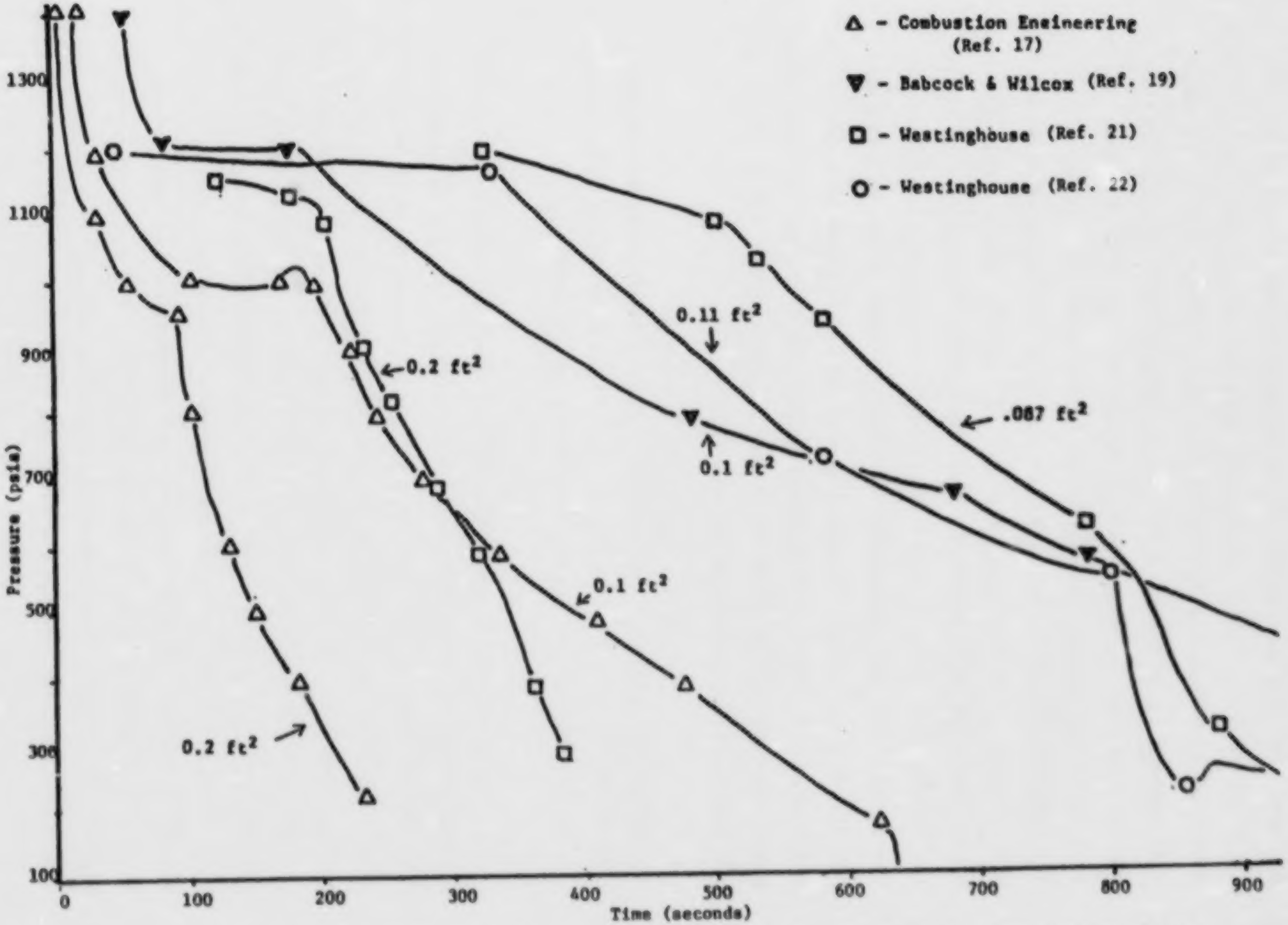


Figure 12 - Fluid Pressure in Core Region vs Time for Various Break Sizes - Pump Discharge Break

sizes. Since the total primary system volumes are similar for the three plants (12,000 ft^3 ± 550 ft^3), and initial fluid conditions are similar, it would be expected that system pressure declines would be very similar for the three plants during a LOCA. This should be especially true for small breaks since the minor differences in pressure drops that exist in the primary system due to different design details would be minimized because of the very low flow rates. However, as illustrated in Figure 12, very substantial differences exist in the core pressure history during the accident. Considering the 0.1 ft^2 plots, along with the 0.087 and 0.11 (the nearest published calculations to 0.1 ft^2), the difference in time from the fastest calculated blowdown to the slowest ranges around 400 seconds, a very large discrepancy. For the 0.2 ft^2 break, comparing the only two plots available, a difference of nearly 120 seconds exists in the time to reach 900 psia, with the difference becoming larger (nearly 400 seconds at 200 psia) as the pressure decreases.

(3) Cladding Temperature - From these very large discrepancies illustrated in Figures 9 through 12, it would be expected that the clad hot spot temperature results would be quite different.

Figure 13 shows these results and, as expected, there is considerable variation. The temperatures plotted are the maximum calculated for the entire spectrum of small breaks by each vendor. They vary from 1075 to 1743°F. The two values by Westinghouse for the 0.087 ft^2 break correspond to two different calculational techniques. The higher temperature (1400°F) corresponds to a pre-May 1974 calculation utilizing the one-primary-system-volume SLAP code. The more recent Westinghouse technique, using the WFLASH code with 16 nodes, shows the maximum at the same break size, but predicts a lower value (1150°F). The two Combustion Engineering points correspond to a similar change in nodalization, with the important exception that the high value (1743°F)

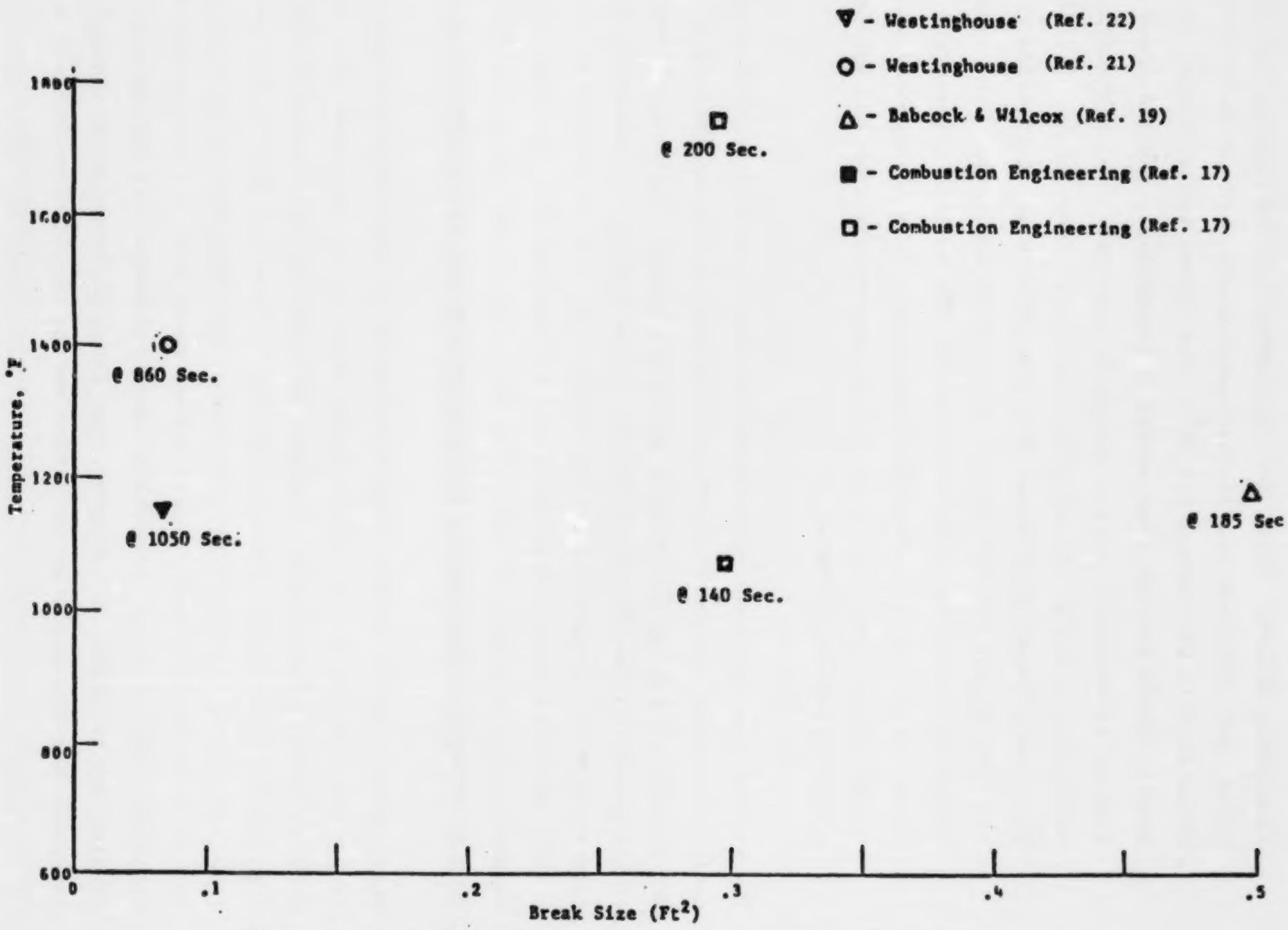


Figure 13 - Peak Clad Temperatures vs Break Size at Pump Discharge Break

corresponds to a multi-node analysis (CEFLASH with more than 15 nodes) used by Combustion Engineering since March 1973, while the lower temperature (1075°F) corresponds to a single node calculation using WALSOB. Thus, for the Combustion Engineering calculations, not only does the maximum temperature occur at a different break size (0.087 versus 0.3 ft^2), the values are different (1150 versus 1743°F for the most recent calculations), and the trend from multi-noding to single noding is reversed. The B&W maximum occurs for a 0.5 ft^2 break with a predicted temperature of 1120°F. Under each temperature value the time after break initiation that the temperature was calculated to occur is indicated. The trend, though not entirely consistent, is for the maximum temperature to occur earlier with increasing break size. This is expected since larger break sizes result in faster depressurization and earlier water loss from the core.

The reasons for the large differences in results described in the preceding discussion cannot be definitely established since details of the various analytical techniques are proprietary. Portions of some of the discrepancies may be attributed to variations in operating design parameters, assumed operating conditions, and variations in the design and operational behavior of the ECC systems for the three plants. However, these differences are quite small and do not appear to provide a reasonable explanation for the substantial variations.

Conclusions - Based on the large differences in results as calculated by the PWR vendors for the small break LOCA, it is concluded that the WASH-1400 assumption (based on these calculations as accepted by AEC-Regulatory) of adequate core cooling for all small breaks if ECC systems work is not sufficiently justified. The fact that in none of the calculations do core temperatures become great enough to cause damage which may lead to melting provides some confidence that adequate core cooling may be achieved. However, the large differences in results indicate that the processes occurring during the accident are not well understood, and that significant effects may be overlooked or improperly considered. In view of the fact that the small break LOCA is a dominating contributor to public risk from PWRs, it is imperative that a substantial justification be provided to establish that ECC systems are adequate for small break LOCAs. In addition, a systematic analysis to demonstrate that ECC protection exists for all small break locations has apparently not been provided for the three PWRs. It is essential that such an analysis be provided to assure that ECC protection is afforded for all small break locations. Such an analysis is mandatory in order to provide independent assurance that the ECC protection for small breaks does in fact exist and that the conservatism claimed for the calculations can be substantiated.

7. PWR Loss of Power Transient Accident Sequence

This accident sequence is described in general terms, applicable to all PWR transients, in Appendix I (Section 4.3.1) of WASH-1400. A more specific consideration of the PWR loss of power transient accident is given on pages 64 through 67 of Appendix V. In this latter description, the significance of this accident on the PWR radioactive release spectrum is stressed. An examination of Table V-16 (Appendix V) reveals that this transient is the dominant contributor to the Category 2 release for PWRs. From Table VI-20 of Appendix VI, it is evident that PWR release Category 2 has one of the highest probabilities of occurrence, and results in the highest number of acute fatalities of any release category. As will be seen, this category contributes to over half of the calculated acute deaths per year for PWRs. Thus, the PWR loss of power transient accident is a significant contributor to PWR accident risks as calculated in WASH-1400.

The accident is initiated by a loss of off-site electrical power. If offsite power is not restored within one-half hour, and the onsite emergency power sources fail, heat removal from the primary system essentially ceases after the steam generators boil dry. Since core decay heat (scram is assumed to occur when power is lost) continues to be transferred to the primary system water, the primary system pressure rises until relief valves open. The water in the core region turns to steam, heat transfer from the core decreases, and eventually the core approaches melting. Melting can occur, in which case the molten core is assumed to fall into the lower vessel plenum after sufficient melting has occurred. Vessel melt-through then is anticipated, followed by melt-through of the containment base material. According to WASH-1400 (Appendix I), containment failure can be caused by four events during this accident sequence:

(1) Accumulation and explosion of hydrogen in the containment from metal-water reaction in the core (designated γ).

(2) A steam explosion in the lower vessel plenum which drives the vessel head into the containment causing failure (designated α).

(3) Excessive pressure as a result of evolved gases produced by decomposition of the concrete base mat of the containment from heating by the molten core material (designated δ).

(4) Melt-through of the concrete base mat (designated ε).

The probability of the accident sequence described, independent of the containment failure mode, is calculated in WASH-1400 (Appendix V, page 66) by evaluating the symbols:

TMLB'

where

T is the probability of loss of offsite a-c power = $2\text{x}10^{-1}$.

M is the probability of nonrecovery of offsite power in 1/2 hour (after which heat removal from the primary system ceases) = $2\text{x}10^{-1}$.

L is the probability of failure of the auxiliary feedwater system - $1.5\text{x}10^{-4}$.

B' is the probability of nonrecovery of offsite and onsite a-c power sources within 1/2 to 1-1/2 hours after the transient event = $5\text{x}10^{-1}$.

The probability of the sequence TMLB', then, is:

$$(0.2)(0.2)(1.5\text{x}10^{-4})(0.5) = 3\text{x}10^{-6}$$

In assessing the probability of the various modes of containment failure, the following values are computed in Appendix VIII of WASH-1400:

$\gamma = 0.13$

$\alpha = 10^{-2}$

$\delta = 3\text{x}10^{-2}$

$\varepsilon = 0.8$

Thus, the probability of a PWR loss of power transient accident with various containment failure modes are calculated to be

TMLB' - $\alpha = 3\text{x}10^{-8}$

TMLB' - $\delta = 1\text{x}10^{-7}$ (rounded from $9\text{x}10^{-8}$)

TMLB' - $\gamma = 4\text{x}10^{-7}$

TMLB' - $\varepsilon = 3\text{x}10^{-6}$ (rounded up from $2.4\text{x}10^{-6}$)

These values are listed in Table V-16, Appendix V of WASH-1400, and assigned radioactive release categories depending on the calculated release from each sequence as assessed in Attachment 1 to Appendix V.

There appears to be a potential mode of containment failure overlooked in WASH-1400 during a loss of power accident. During the accident, the primary system pressure will remain at or near the set point of the pressure relief valves (above the operating pressure of ∿2200 psi). The superheated steam generated in the core region will maintain the system pressure. When the core eventually melts through the bottom of the reactor vessel (assuming no steam explosion in the vessel), the primary system pressure will be rapidly relieved (see also Comment No.1 under General Comments at the end of this section). The molten material will be spewed into the reactor vessel cavity, and the steam (and water which flashes to steam) in the primary system will rapidly (within a few seconds) exit the reactor vessel into the containment causing an abrupt pressure rise. As the primary system pressure rapidly decays to about 600 psi, the accumulators will begin discharging cold water into the reactor vessel downcomer. This water will flow down the reactor vessel and directly out the hole created by the core melt-through. The water will impact the molten mass in the reactor vessel cavity. Large amounts of energy, in the form of steam, will be generated and released to the containment, causing an additional loading just seconds after the primary system blowdown. Calculations have not been performed, but it is possible that this second abrupt pressure loading will fail the containment, especially in the absence of the operation of containment heat removal systems, which must be assumed to be disabled due to the loss of electrical power. In addition, a steam explosion may be more likely under these conditions, as discussed below, than when the molten core contacts water in the lower plenum of the reactor vessel. Neither of these potential containment failure modes is considered in WASH-1400.

Little is known about the thermodynamics of explosions when water comes into contact with molten material. Numerous examples of explosions have occurred under these conditions, some of which are described in Appendix VIII (Appendix B) of WASH-1400. Appendix VIII (pages 27-29) also derives an expression for the probability of a vessel steam explosion which causes containment failure by blowing the vessel head against the containment. The formula used to compute the probability is

$$P_1 = P_{fw} \alpha_{sxs} \alpha_{cf}$$

where

P_{fw} is the probability that the fuel will come in contact with water. This factor considers the possibility that the fuel will become vaporized (and not available to cause the explosion) or that there will be no water left in the lower plenum when the molten core falls.

α_{sxs} is a term which describes the likelihood that the explosion will occur if the molten core comes into contact with the water. This term includes the empirical observation that saturated water, which will exist in this case, does not appear to be nearly as effective in causing steam explosions as subcooled water. The term also includes a consideration that the fuel may not disperse in the water sufficiently to cause an explosion. The value of α_{sxs} is largely a matter of judgment based on observations from various sources of molten material-water interations.

α_{cf} is the probability that the vessel head, following the steam explosion, will cause rupture of the containment.

These quantities are assigned the following values in WASH-1400, with large uncertainty bounds,

$$P_{fw} = 0.89$$

$$\alpha_{sxs} = 0.1$$

$$\alpha_{cf} = 0.1$$

which yields a (rounded) value for P_1 of

$$P_1 = 0.01$$

This value is used for α, as described previously in WASH-1400.

To assess the probability of containment failure under the conditions of rapid primary system blowdown at the time of vessel melt-through, followed by impingement of accumulator water on the molten core mass in the reactor vessel cavity, it is necessary to reconsider the terms used to compute α. In this case P_{fw} is probably greater than the WASH-1400 value of 0.89 since water availability is assured by the accumulator discharge. Thus, $P_{fw} = 1$ appears a reasonable, if slightly conservative, value. The value of α_{sxs} should be raised since:

(1) subcooled water from the accumulators will come into contact with the molten core mass, whereas saturated water is appropriate for

the case assumed in WASH-1400. Subcooled water in contact with molten material enhances the probability of an explosion. Although some heating of the accumulator water will occur as it flows through the inlet pipes and the vessel downcomer, the short path lengths involved and the high volumetric flow rate caused by the high pressure drop between the accumulator tank and the primary system will probably minimize this effect. Appendix VIII (Page 49) states that the probability of a steam explosion when molten fuel drops into subcooled water is 0.5 versus 0.01 for the saturated water case.

(2) Intimate contact between the accumulator water and the molten mass in the reactor cavity will likely occur because of the impingement of the accumulator water on the molten mass. Also, some dispersion of the molten material is likely due to the large area of the reactor vessel cavity floor. Both of these effects will enhance the probability of a steam explosion over the case considered in WASH-1400. In view of these considerations, and in light of the uncertainty involved, it seems appropriate to assign a value of 1.0 to this quantity.

The term α_{cf}, which accounts for the likelihood of containment failure from impingement of the reactor vessel head on the containment dome, is probably the most uncertain of all the factors. For the case of the reactor vessel cavity steam explosion, the pressure pulse which accompanies the explosion will be superimposed on the containment pressure already existing from the rapid primary system blowdown occurring at the time of vessel melt-through. This will tend to increase the probability of containment failure from overpressure.

It is impossible, without further analysis beyond the scope of this effort, to determine the extent to which α_{cf} should be increased for the reactor vessel cavity steam explosion. In order to maximize the sensitivity to this parameter, and to conservatively account for the uncertainty, a value of 1.0 was assumed.

Table 15 summarizes the changes which would occur in the WASH-1400 analysis from the considerations of the preceding discussion.

Table 15 - REVISED PROBABILITY VALUES FOR PWR LOSS OF POWER TRANSIENT WITH CONTAINMENT FAILURE

Parameter	P_{fw}	α_{sxs}	α_{cf}	α/α'*	TMLB'	TMLB'-α/α'
WASH-1400	0.89	0.1	0.1	0.01	$3x10^{-6}$	$3x10^{-8}$
Revised	1.0	1.0	1.0	1.0	$3x10^{-6}$	$3x10^{-6}$

* α is the WASH-1400 containment failure mode consisting of the reactor vessel head impacting on the containment dome as a result of a steam explosion in the lower reactor vessel plenum; α' is the containment failure mode consisting of the reactor vessel head impacting on the containment dome or containment overpressure, either of which result from a steam explosion in the reactor vessel cavity.

Table 16 is a revision of a portion of Table V-16, Appendix V, WASH-1400. The table shows the contribution to release categories 1 and 2 of the PWR loss of power transient (TMLB') as described previously. The sequences show three modes of containment failure: α (vessel steam explosion), γ (hydrogen explosion), and δ (overpressure from gases evolved due to concrete base mat heating). The additional mode of containment failure described previously is shown as TMLB'-α'. This sequence has been somewhat arbitrarily placed in release category 2. A precise determination of its proper release category placement would entail a rather complex analysis of the fission product release associated with this new accident sequence. However, it should fall into Category 1 or 2 since it is similar to, and chronologically between sequences TMLB'-α and TMLB'-δ. Its placement in Category 2 will maximize its influence on the PWR risks.

Table 16 - COMPARISON BETWEEN WASH-1400 and REVISED PWR RELEASE CATEGORY 1 and 2 PROBABILITIES

	RELEASE CATEGORIES	
	1	2
Transient Event - T	TMLB' - α 3×10^{-8} (3×10^{-8}) *	TMLB' - α 4×10^{-7} (4×10^{-7}) TMLB' - δ 1×10^{-7} (0) TMLB' - α' 0 (3×10^{-6})
T-Probabilities	9×10^{-8} (9×10^{-8})	5×10^{-7} (3.4×10^{-6})
Summation of all accident sequences per release category		
Median	7×10^{-7} (7×10^{-7})	5×10^{-6} (7.9×10^{-6})

* Numbers in parentheses are revised values as described in the accompanying discussion.

The numbers in parentheses in Table 16 are revised figures resulting from consideration of sequence TMLB'-α'. Since the probability of α' (containment failure from accumulator water impacting the core melt in the reactor vessel cavity) has been assigned a probability of one, the release contribution from TMLB'-δ must be eliminated (reduced to zero) since it would occur after the α' event. As shown in the table, the median value for the summation of all accident sequences per release category for Category 2 is raised from $5x10^{-6}$ to $7.9x10^{-6}$. To determine the significance of this increase on the average acute fatalities per year from PWR accidents, the appropriate parts, with revisions, of Table VI-20 (Appendix VI) from WASH-1400, are shown in Table 17. The average acute fatalities per year are determined by the product of the accident probability per year and the average acute fatalities for each release category. The increase in Category 2 as a result of the sequence TMLB'-α', as shown in Table 17, is from $3.1x10^{-4}$ to $5.0x10^{-4}$. This results in a total increase, from all accident sequences which cause acute fatalities, of from $5.4x10^{-4}$ to $7.3x10^{-4}$, a 35 percent increase. (It should be noted that, using the WASH-1400 technique of adding 10 percent from adjacent categories, Categories 1 and 3 will be increased as a result of the increase in Category 2. However, as can be seen in Table 17, the increase in average acute fatalities per year will not be significant.)

Conclusions - The possibility of containment rupture from either overpressurization or damage from vessel head impact as a result of a steam explosion or generation when accumulator water impacts the molten core mass in the reactor vessel cavity following a loss-of-power transient accident seems to have been overlooked by WASH-1400. A somewhat conservative evaluation of the effect of this accident indicates that the average acute fatalities per year from all PWR accidents could increase by as much as 35 percent. Further analysis is required to firmly establish the effect of this change.

Table 17 - COMPARISON BETWEEN WASH-1400 AND REVISED AVERAGE ACUTE FATALITIES PER YEAR FROM PWR LOSS OF POWER TRANSIENT ACCIDENT

	PWR Release Category	Accident Probability Per Year	Average Acute Fatalities	Average Acute Fatalities Per Year
WASH - 1400	1	7×10^{-7}	34	2.4×10^{-5}
	2	5×10^{-6}	62	3.1×10^{-4}
	3	5×10^{-6}	39	2.0×10^{-4}
	4	5×10^{-7}	2.7	1.4×10^{-6}
	5	1×10^{-6}	.22	2.2×10^{-7}
			WASH-1400 Total = 5.4×10^{-4}	
REVISED	1	7×10^{-7}	34	2.4×10^{-5}
	2	8×10^{-6}	62	5.0×10^{-4}
	3	5×10^{-6}	3.9	2.0×10^{-4}
	4	5×10^{-7}	2.7	1.4×10^{-6}
	5	1×10^{-6}	.22	2.2×10^{-7}
			Revised Total = 7.3×10^{-4}	

Miscellaneous Comments - In reviewing the WASH-1400 analysis of the PWR loss-of-power transient accident, the following comments were developed:

(1) Appendix VIII, page A-32 to 34 - The analysis of reactor vessel melt-through by the molten core as described here appears to apply only to the LOCA case where essentially no pressure exists in the primary system at the time of melt-through. For transient accidents, the primary system pressure can be as high as the pressure relief valve settings (in excess of 2200 psi) at the time that the molten core begins to heat up the lower reactor vessel head. This pressure could accelerate reactor lower head failure and alter the chronology of the accident sequence from that presented in WASH-1400 (Appendix V) which uses the Appendix VIII analysis.

(2) Appendix VIII, Table B-2, page B-10 - It is not clear what the column labelled "T,C" represents.

(3) Appendix V, page 62 - The assumption is made here that a transient accident which results in failure of reactor coolant system relief and safety valves to close becomes a small LOCA and the PWR small LOCA event trees were considered to be applicable. It should be noted that assumptions in WASH-1400 relative to the small break LOCA sequence and requirements for core protection are based on vendor calculations of the accident. These calculations are made assuming steady state conditions exist prior to the accident (see, for example, page 15.3-3 of Reference 2.2). A transient accident creates off-normal core thermal and fluid hydraulic conditions (see Reference 23). These differences could alter both the sequence of the accident as well as requirements for core protection.

8. BWR-PWR Component Failure Modes and Rates

This Section presents the results and conclusions of the review of Appendices III (Failure Data), IV (Common Mode Failures) and X (Design Adequacy) of WASH-1400. In reviewing these appendices, the following questions were considered:

(1) Are the data sources used in Appendix III applicable and properly applied?

(2) Are there other applicable data sources which were not used by WASH-1400?

(3) Have common mode failures been properly considered and accounted for in Appendix IV?

(4) Has appropriate attention been given to consideration of design adequacy, including environmental effects, aging, etc?

(5) Are the component failure rate values properly computed from the data used by WASH-1400?

The findings related to the above questions for each Appendix are as follows:

Assessment of Failure Data (Appendix III) - The component failure rate assessment of median, upper and lower bounds, and the error factor based on data in Table III-1 does not appear to be consistent. The credit taken for nuclear design and fabrication is also not adequately explained or consistently applied. Examples of deficiencies in failure rate assessment are:

Pipe rupture data, pipes >3 inches - The portions of Appendix III pertinent to the derivation of large pipe rupture probabilities are widely scattered, inconsistent and difficult to reconcile. Units for pipe failures are mixed, and not readily converted; it is not clear which data values pertain to "LOCA sensitive piping"; it is difficult to determine which failure rates apply to nuclear, non-nuclear, nuclear processing, etc, piping; the definitions of pipe "failure", "rupture", "complete rupture", "complete severance" are not given although these terms are each used; and it is not apparent which data apply to which pipe size, since several different pipe size categorizations are used.

Table III-1 presents the results of the data survey in which pipe ruptures are presented for "Hi Quality" (not defined) pipe greater than 3 inches in diameter. The units used in Table III-1 are ruptures per section per hour. The data in Table III-1 are plotted as Figure 14 (log-normal paper) using a method suggested by Gumbel(24) and Ferrell(25) which merely adjusts the position of the data points slightly to account for the fact that other data may exist which fall near the extremes. A best fit curve has been drawn through the data and the WASH-1400 curve, using the upper, lower and median values given in Table III-1, and is shown for comparison. The circled values are apparently derived from nuclear experience, as indicated by the headings in Table III-1 and the descriptive material provided for the Table III-1 references on pages 191-199. As can be seen, the WASH-1400 assessed range is substantially lower (two orders of magnitude at the median value) than the best fit curve. The reasons for this disparity are not apparent, even though section 2.1 (page 18) states that the Table III-1 values "...formed the bases for the assessed ranges."

Table III-3 (page 26) lists a comparison of WASH-1400 assessments with industrial experience. Under the "Active Mechanical Hardware" portion of the table, pipe "Plug/Rupture" failures are listed. While it is not clear why "plug" (plugged pipe) failures were included in this table and not in Table III-1, the assessed upper and lower bounds for greater than 3-inch diameter pipe correspond to the values given in

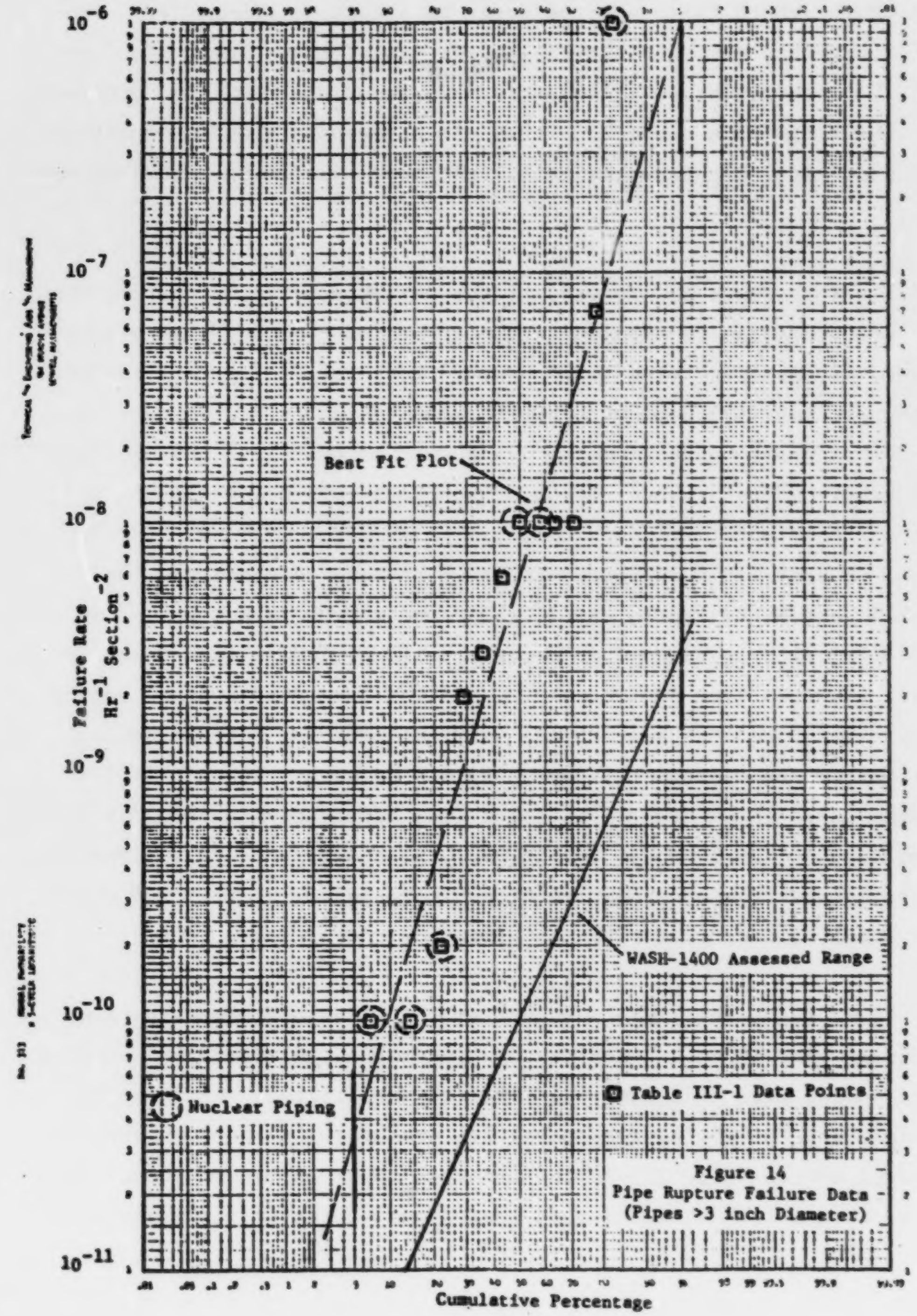


Figure 14
Pipe Rupture Failure Data
(Pipes >3 inch Diameter)

Table III-1. However, the lower bound listed for industrial experience is 3×10^{-10}/hr and there is no corresponding value from any source in the Table III-1 tabulation even though Section 2.2 (page 18) states that the Table III-2 values are "extractions" from Table III-1.

An examination of 1972 nuclear experience is presented in Section 3, page 29. Table III-6 (page 38) contains piping failures based on 11 failures for 1972, and a failure rate of 1×10^{-9} per hour per foot is derived. Conversion of this rate to the units used (ruptures/section/hr) in Table III-1 is difficult. The rate would be 8.76×10^{-6} failures per year per foot. Page 184 states that the amount of pipe per plant is "taken as roughly 170,000 feet." The amount of pipe per section varies. Page 101 states that a pipe section is "approximately 10 to 100 feet." Thus, there are, very roughly, 1700 to 17,000 pipe sections per plant. The Table III-6 pipe failure rate becomes 1.5×10^{-2} to 1.5×10^{-1} failures/section-year with a large uncertainty resulting from conversion uncertainties. However, since it is not clear (1) what is meant by a pipe "failure" for the nuclear experience data in Table III-6, (2) what pipe sizes the nuclear data represents, or (3) which piping on the plants was considered in the data, a comparison between the Table III-1 and III-6 values is probably not valid.

Another problem appears to exist in Table III-6 which lists 280,000 feet as the total length of piping considered in the survey for eight PWRs and 315,000 feet as the total length for nine BWRs. This produces a pipe length per plant of 35,000 for both types. This does not agree with the 170,000-foot value quoted on page 184 of Appendix III.

Section 6.3 (page 174) provides a discussion of how LOCA initiating pipe rupture rates were derived for used in WASH-1400. For pipes larger than 6 inches, a rate of 1×10^{-4} per plant per year is listed. Substantial discussion is included in support of this rate, and the rate appears to be reasonably well justified in spite of the problems discussed previously. However, in the Nuclear and Nuclear-Related

Experience subsection (page 178), reference is made to the "1972 nuclear history examined for the general data base." Ten failures are listed, while the 1972 nuclear data in Tables III-5 and III-6 each list 11 failures. It is not clear why these numbers are different.

Pipe Rupture Data, Pipes <3 inches - Many of the deficiencies listed previously for the Appendix III discussion of pipes greater than 3 inches exist for pipes less than 3 inches. Figure 15 shows a comparison between the WASH-1400 assessment and a best fit of the data given in Table III-1. Again, the WASH-1400 assessed range falls below a best fit to the data. As before, the apparent reason is to account for the improved failure rate for nuclear piping, yet the nuclear experience data points as indicated on Figure 15 do not substantiate the assumed improvement. According to Table III-1, the U.S. nuclear experience gives a value of $1x10^{-9}$, which was assumed to be the median value in the WASH-1400 assessment. However, the nuclear experience value given in Table III-2 is $2x10^{-9}$ for pipes less than 3 inches. It is not clear why these numbers are different.

A further area of confusion is introduced in Tables III-2 and 3 where pipe plugging data appear to be included for all pipe sizes. In Table III-10 (Summary of Assessment for Mechanical Hardware), a plug failure is said to apply only to pipes <3 inches. A plugged pipe would usually result in an entirely different accident (not a LOCA) than a ruptured pipe, and it is not clear why the two are combined since the results are used for LOCAs. Also, the pipe failure mode for the nuclear data compiled in Section 3.2 is not identified.

A further area of confusion exists relative to pipe size. Table III-1 classifies pipe as either greater or less than 3 inches; the pipe failure data discussion in Section 5.4 of Appendix III classifies pipe failures as initiating events for LOCAS in three size categories - 1/2- to 2-inch, 2- to 6-inch and greater than 6 inch. Page 175 of Section 6.4 indicates that the pipe rupture data used in WASH-1400 could be broken into two classes, either greater than 4 inches or less

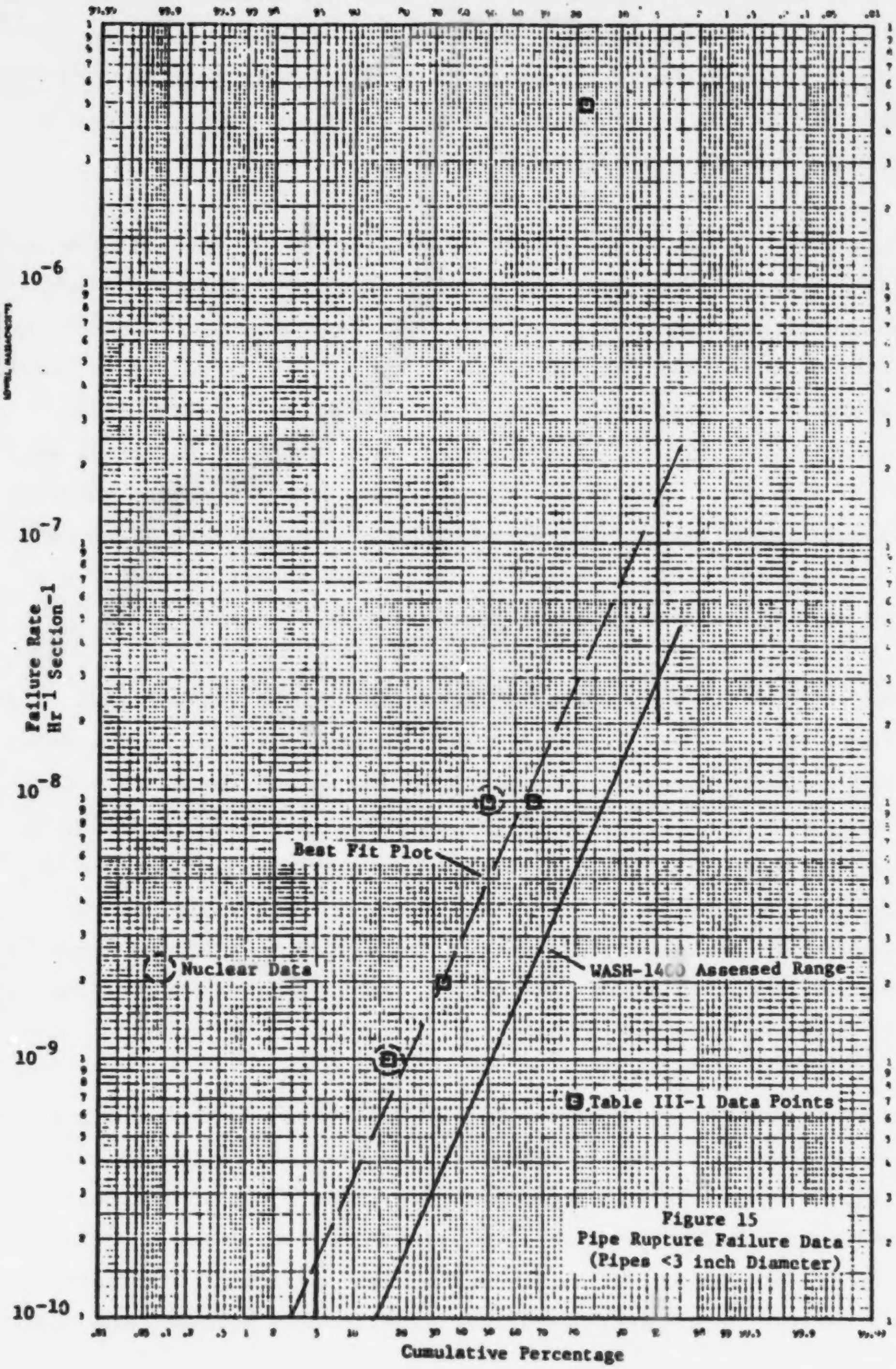


Figure 15
Pipe Rupture Failure Data
(Pipes <3 inch Diameter)

than 4 inches. It is not clear how the data were used to obtain failure rates depending on pipe size. Further confusion is generated in Section 6.4 by citing numerous pipe failure data sources, some of which are not apparently listed in Table III-1, although part of the problem may be due to typographical and reference number errors in the source headings of Table III-1.

Check valve failures - The tabulated description of this failure mode in Table III-1 is "reverse leak", yet in Table III-10, which summarizes the WASH-1400 assessments, the failure is described as "internal leak (severe)." (The same failure value is used.) It is not clear which failure mode is being considered - any leak or only severe leaks. (Presumably, reverse leak and internal leak for check valves is equivalent.) In view of the rather significant nature of severe reverse leakage in check valves determined elsewhere in WASH-1400 (see Section 4.4, Appendix V), a detailed discussion of what constitutes a severe reverse leakage in check valves should be provided.

Relief valve failure-to-reseat - The relieve valve failure mode, "Failure-to-reseat", is not assigned a rate in Table III-1. Yet, a value of 10^{-1} to 10^{-2} is cited ("estimated on an engineering basis") for this mode on pages 62 and 69 of Appendix V. This mode should be included in Table III-1 and its failure rate derivation should be discussed in Appendix III. Apparently, data are available for this failure mode since the valve malfunctions tabulated for 1972 on pages 54 to 59 of Appendix III show that 5 of 13 relief valve malfunctions were of the failure-to-reseat mode.

Pumps failure-to-run - A log-normal plot of pump failure-to-run rates from data sources given in Table III-1 is shown in Figure 16. As can be seen, 13 of the 14 data points for failure-to-run in a normal environment fall below the WASH-1400 assessed median value of $3\text{x}10^{-5}\text{hr}^{-1}$, apparently indicating, in this case, nuclear experience indicated a higher failure rate than non-nuclear experience. However, the U.S.

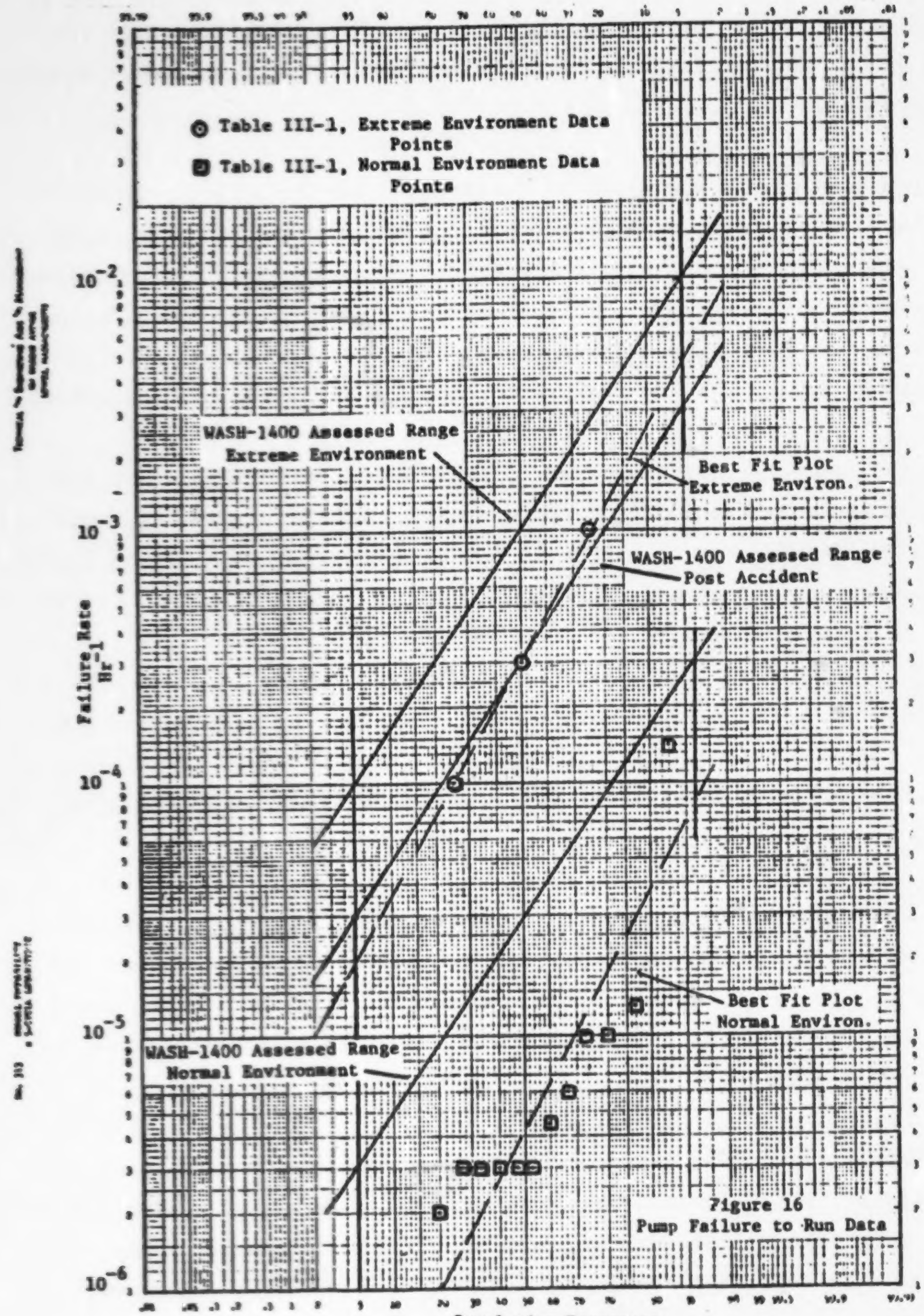


Figure 16
Pump Failure to Run Data

Nuclear Operating Experience column of Table III-1 gives a value of $3\times10^{-6}\text{hr}^{-1}$. (In many other cases in the Table, the value in this column was used for the assessed median.) It appears inconsistent to use a higher value, particularly when the non-nuclear data argue for a lower assessment.

Three data points are given in Table III-1 for failure-to-run under a severe environment. A best fit plot of these data, shown in Figure 16, is essentially equivalent to the WASH-1400 assessed range for failure-to-run after recovery of the post-accident environment and a factor of two to five below the assessed range for severe environment before post-accident recovery. The reasons for these differences are not clear.

Diesel generator failure-to-start - The data points used in Figure III-8 have been replotted on log-normal probability paper as shown in Figure 17. The values suggest a negative skew to the distribution of failure rates, and indicate either that the sample selection was poor or that log-normal distribution is inappropriate.

When reviewing the other data contained in Table III-1, it is found that the plotted points (Figure III-18) represent three out of five data points with the two omitted data points being much lower than the plotted points.

It appears that the range of failure-to-start rates as assessed by WASH-1400 for diesel generators is too narrow. It also appears that the range of rates tabulated and the resulting plot given in Figure III-18 indicate that the proper range cannot be established from the available data.

An investigation was made of diesel generator failure rate sources. Referring to the failure rate tabulations for diesel generator failure rates in Table III-1, particularly for failure to start, it was found that important sources of data referenced in the AEC Report

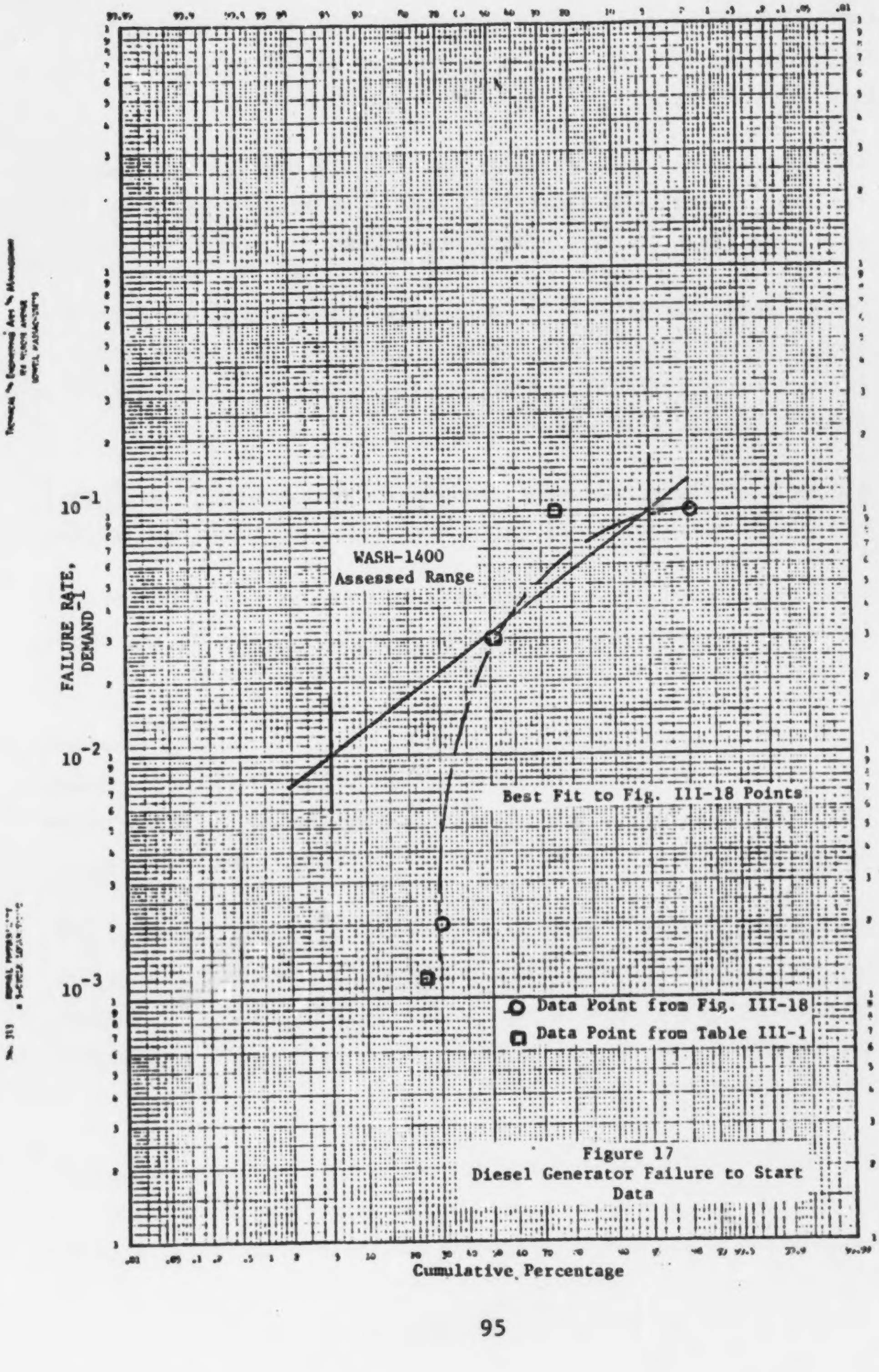


Figure 17
Diesel Generator Failure to Start Data

OOE-OS-002(26) were not listed. In addition, the data cited in Table III-1 for the UKAEA Systems Reliability Service (SRS) appear to be high at least a factor of ten when compared with the SRS reference in OOE-OS-002 and to the value given in the SRS Report SRD-R-16(27). Specific examples are:

(1) UKAEA statistics referenced in Section 4.4 of OOE-OS-002 show a failure-to-start probability of $1x10^{-2}$ to $2x10^{-2}$ per demand, not $1x10^{-1}$ as given in Table III-1, WASH-1400.

(2) UKAEA Report SRD-R-16(27), Table III, lists a failure-to-start probability of $2x10^{-2}$ per demand, not $1x10^{-1}$ per demand and a failure to run rate of 0.2 per year or $2x10^{-5}hr^{-1}$, not $1x10^{-3}$ as given in Table III-1, WASH-1400.

(3) A U.S. Army study referenced in Section 4.4 of OOE-OS-002 gives a computed probability of failure-to-start as $7x10^{-3}$ per demand.

(4) A General Motors report referenced in Section 4.4 of OOE-OS-002 gives a failure-to-start probability of $1x10^{-2}$ per demand for GM units in service as peaking units from 1957 to 1969.

(5) Statistics compiled from U.S. nuclear power plants and reported in Table VII of OOE-OS-002 establish a failure-to-start probability ranging from $1.5x10^{-2}$ to $3.4x10^{-2}$ per demand. The range reflects variations in manufacturer and capacity of the units.

In addition to the preceding data, the Edison Electric Institute(28) provides failure-to-start data for 203 diesel generator units representing 806 unit years of operation. These data establish a failure rate for starts of approximately $9x10^{-2}$ based on 107,679 successful starts in 118,563 attempts.

The data listed in Table III-1 for diesel generator failure-to-start and failure-to-run (complete plant) should be revised to include the preceding data. A log-normal plot of failure-to-start data, including the preceding data, is given in Figure 18. This plot is based on the data points from Table 18.

Table 18 - DIESEL GENERATOR FAILURE-TO-START DATA

Source	Failure-to-Start/Demand
UKAEA (SRS)	2×10^{-2}
U. S. Army	7×10^{-3}
General Motors	1×10^{-2}
USAEC (OEE-OS-002)	3×10^{-2}
LMEC (Table III-1)	1.2×10^{-3}
IEEE Trans (Table III-1,	1×10^{-5}
NRTS (Table III-1)	1×10^{-8}
EEI Pub 74-57	9×10^{-2}

The tabulation in Table 18 omits U.S. and UKAEA data from Table III-1 which are probably covered by the data in the tabulation. The data are plotted using the method of Gumbel and Ferrell.

It appears, based on Figure 18, that WASH-1400 has used a median value three times larger than suggested by the larger collection of data and that the error factor associated with the revised data is 30, not 3 as shown in WASH-1400. The overall conclusion is that the WASH-1400 assessed data may be unnecessarily conservative.

Human Reliability Analysis (Appendix III) - The final human reliability assessments used in WASH-1400 are described as being values adjusted from the basic "General Error Rate Estimates" given in Table III-13. While the factors considered in adjusting these basic rates appear appropriate, as qualitatively described following Table III-13 (page 131), very little in the way of quantitative assessments are presented. Referring to these basic error rates as being "lower", "higher", or "markedly reduced", is quite vague.

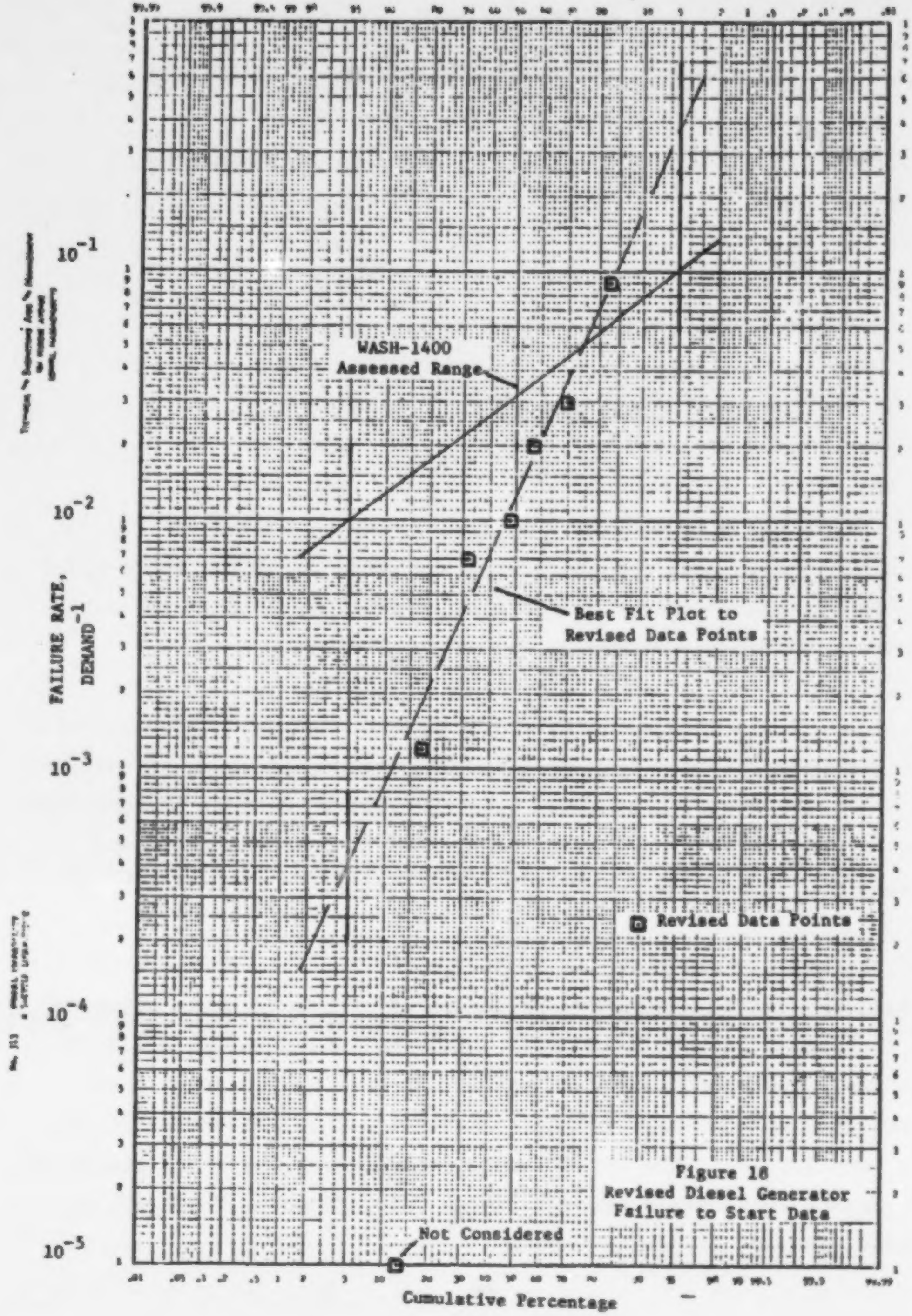


Figure 18
Revised Diesel Generator
Failure to Start Data

The human error contribution values appear explicitly in the fault tree analyses in Appendix II (Vols. 2 and 3). These values are described as being derived in Appendix III. However, it is not possible to trace the origin of these values in Appendix III except through the very general discussion of how the basic rates were adjusted. It would be helpful, as a minimum check on consistency, if the values used in Appendix II also appeared as an extension of Table III-13. As noted elsewhere in Section V of this report, some human error contributions appear to be incorrectly and inconsistently assessed.

The example analysis given in Section 6.1.4 of Appendix III is not at all convincing that all *plausible* modes of operator action under emergency conditions have been evaluated or that the probability of overlooking an important failure path has been decreased to a negligible value. Specific reasons for this observation are:

(1) The example given is not complete. The discussion of recovery steps on pages 152 and 153 of Appendix III is not illustrated in Figure III-12. The selection of incorrect valves other than MOV-1863A and B has not been developed. The selection of incorrect valves other than MOV-1863A and B *will not* produce a recovery situation as described on page 152. The selection of valves other than 1863A and B leads to many other ensuing sequences for recovery or failure which are not considered. However, if it is assumed that no immediate recovery occurs, then an additional failure probability of 0.999 (10^{-2}) (0.25) $\simeq$ 2.5×10^{-3} is identified for failing to accomplish step 4.8.1. This added probability increases the value estimated in the example by a factor of three.

(2) The conclusion that *all* three people in the control room would recognize that the RWST level had reached 14.5 percent and that action is required seems highly optimistic (page 150). Each operator would likely be quite busy and would not check the other

two to see that they are performing correctly. In addition, it is possible that only one operator would be reading the written instructions and the others would not necessarily be aware of the significance of and action required by the various gauges and alarms.

(3) The rationale for assuming an error rate of 10^{-2} for selecting the wrong valve control switches rather than the basic error rate of 10^{-1} appears arbitrary as presented in the last paragraph on page 151. Though the rate selected may be appropriate, it would appear that additional rationale is required, rather than simply saying that 10^{-2} was picked because the previously selected "basic" value of 10^{-1} seemed too large in this situation.

(4) The example does not clearly identify the failure being evaluated. From reading the discussion, it appears that the event of interest is failure-to-open MOV-860A and B. Lack of a tightly developed statement of the problem, approach, and results in this example seriously diminishes its value. The example should be revised to identify clearly the event of interest.

(5) The example discussion does not indicate how much time is available after making a wrong decision to correct the error. The greater the length of time available, the more chance there is for error feedback to lead to error recovery. Of course, with longer times required to recover the error, the magnitude of consequences may increase. Hence, there must be a time limit for which error recovery has no real value in consequence limitation. This point is not discussed. The errors and questionable assumptions in this example do not lend confidence that human reliability factors were handled appropriately in the many other situations which are assigned values without supporting analysis or discussion.

Miscellaneous Comments (Appendix III) - The following comments were developed during the review of Appendix III:

(1) It would be helpful if the discussion of data treatment on pages 15 to 18 were replaced with a discussion having content similar to that of Section 3.6.1 of Appendix II. The methods of data treatment given in Section 3.6.1 of Appendix II are much more lucid and appropriate than the discussion now contained in Appendix III. In particular, the reasoning associated with use of the log-normal distribution for data range selection is much better developed in Appendix II and belongs also in Appendix III.

(2) Both Appendix II and Appendix III should include a brief discussion to indicate that the log-normal distribution is used to establish a range of failure rate values, while the exponential distribution is used to estimate the probability of failure. It is only by carefully reading the text (page 83 of Appendix II) that it is clear that the exponential distribution has been used for computing failure probabilities.

(3) The failure rate data source given in Table III-1 for Holmes & Narver is incorrect. The report which should be cited is HN-199(29) for the four-month data collection activity at Connecticut Yankee (Haddam Neck). Also, no reference is made to a much larger body of reported nuclear experience data contained in HN-185(30).

(4) The quantity n_t on page 33 should be n_f.

(5) The last two sentences of the second paragraph on page 33 seem inconsistent. Specifically, if any task error rate less than 10^{-5} is viewed with skepticism, then any error rate computed to be less than 10^{-5} should be rounded up to 10^{-5} rather than being dropped as implied in the last sentence and as specifically illustrated in the example in paragraph one on page 151, where an

estimated error rate of 10^{-7} led to the decision to drop the error from further consideration. It would seem more reasonable to assign a value of 10^{-5} to any critical human error which was found by estimate to be less than 10^{-5}.

(6) In the equation at the top of page 172, it appears that the second exponential term, $\exp - \frac{\tau' t_o + 1}{\tau}$, should be $\exp - \frac{\tau' t + 1}{\tau}$

(7) The heading "Valves (SOV)" in Table III-1 appears to be misplaced and should be moved down one row.

(8) In several instances, the WASH-1400 assessed ranges for component failure rates are significantly narrower than would be indicated by the data listed in Table III-1. Two notable examples are "Electrical Clutch Failure" and "Motor Operated Valves Failure-to-Operate."

(9) Page 26, Table III-3 - This table is labelled "Active Mechanical Hardware" and includes pipe failures. Pipes should not be considered "active mechanical hardware."

Common Mode Failures (Appendix IV) - The lack of a clear common mode failure definition in WASH-1400 tends to confuse the treatment of these failures. Common mode failures are defined very broadly in Appendix III, page 40, as "failures having a common cause." Another definition is given in the first paragraph, page 5 of Appendix IV, namely, "multiple failures which are dependent." This latter definition is confusing because it introduces the concept of "dependent" without identifying whether they are dependent on a single external influence, dependent on each other, or both. All failures are dependent on something. Therefore, dependency is not a concept peculiar to common mode failure. Elsewhere in Appendix IV the phrases "dependent and common mode" and "common mode and dependency" are used. Use of dependent or dependency along with common mode clutters the concept since a special case of dependency is implicit in "common mode."

Normally, common mode failure is defined as a single cause which fails more than one redundant item or function. Several investigators have promulgated this definition of common mode failure. For example, W. C. Gangloff(31) uses it in the first paragraph of his introduction. Similarly, Green and Bourne(32) define common mode failures in this manner. The American Nuclear Society also offers a similar concept in Paragraph 3.6.8 of the draft standard N18.8(33). In this document, common mode failures are defined in context as follows:

> "The designer shall recognize that the redundant or diverse channels do not necessarily have independent failure modes. He shall seek to eliminate multiple channel failures originating from a common cause. The failure modes of redundant and diverse equipment, channels, and systems and the conditions or operations that are common to them shall be studied to determine that a predictable common failure mode does not exist."

Finally, the AEC (now Nuclear Regulatory Commission) in Regulatory Guide 1.6(34) has discussed common mode failures in context with the use of redundancy in standby power systems. Here, common mode failures are discussed in conjunction with loss of redundancy through a common failure cause.

Insufficient information is contained in Appendix IV relative to quantification of common mode failures for specific application to the systems analyzed in Appendix II. Although a reasonable qualitative description of common mode failures is given, exactly how these considerations were numerically applied to specific systems does not appear adequate. A related deficiency is that the Appendix does not contain a clear presentation of a systematic approach used in identifying and quantifying common mode failures. A flow chart describing each step taken in such identification and quantification would be very valuable.

Section 5.3, "PWR LOCA Sequence Common Mode Failure Evaluation," especially needs a quantitative evaluation. Only one quantitative value is given. It would be useful if calculated probability were presented for many of the common mode failures discussed. This could be done in tabular form by sequence.

Miscellaneous Comments (Appendix IV) -

(1) General - Sections 5.5.1 and 5.2, as labelled in the table of contents and in the text, are identical to Sections 6, 6.1 and 6.2. The fact that Section 5 pertains to PWRs and Section 6 to BWRs should be identified in these titles. Also, to be consistent, Sections 5.3 and 6.3 should be the same, and a Section 6.4, equivalent to 5.4, should be added.

(2) Page 7, Table IV-1, Item V-2 - While design adequacy is mentioned here and is addressed in Appendix X, neither Appendix X nor Appendix IV identifies the probabilities or frequencies associated with earthquakes or other external events which might give rise to common mode failures. Appendix IV is deficient in this respect. Even if the probabilities of these external events are addressed in other parts of WASH-1400, Appendix IV is the proper place to discuss them quantitatively and qualitatively with respect to their contribution to common mode failure.

(3) Page 44, Last Paragraph - The choice of a log-normal distribution for common mode failure probabilities is used, but no justification is provided other than the previous arguments given for component failure rates. The use of log-normal distributions for common mode failures should be justified by indicating that:

(a) Experience with common mode failures demonstrates the acceptability of the log-normal distribution, or

(b) There is no major difference in results when using any other, possibly acceptable, distribution for the common mode failure distribution.

(4) Section 5.2, Summary of Methods - This section should be titled "Summary of Evaluations." Reference is made to subsections 4.1 and 4.2 on page 70. There are no such subsections.

Design Adequacy (Appendix X) - This Appendix describes the results of an effort performed to determine whether external phenomena and adverse environmental conditions from accidents represent a common mode failure potential. However, the Appendix falls short in its evaluation because there seems to be no conclusion or summary which clearly states which conditions represent a problem, what the magnitude of the effect is, and how such considerations were utilized in the system failure evaluations elsewhere in WASH-1400.

What appears to be required in Appendix X is a systematic probabilistic determination of the likelihood of common mode failures from external events and accident conditions. Such a determination should consider the probability of the external event exceeding the design condition as well as the probability that the design and/or fabrication is inadequate for events less severe than the design condition. This approach would result in a quantified evaluation amenable to incorporation into the general format of the WASH-1400 risk evaluations.

Appendix X indicates that in a significant number of cases insufficient information was available to determine design adequacy. It is not clear how these cases were accounted for in the risk assessments.

The evaluation does not seem to be complete in that only a limited number of equipment and structure selections were made (a "sample" according to page 1). Section 3 does indicate that the selection was biased towards selecting those components and systems which had safety functions and were vulnerable to high stress. However, the Appendix does not indicate how large the sample is in relation to all systems and components of interest.

Conclusions - As a result of the review of Appendices III, IV and X, the following responses have been developed to the questions posed at the beginning of this Section:

(1) Are data sources used in WASH-1400 applicable and properly applied?

The general answer is "yes." There are, however, specific instances cited in the detailed review where application is questionable, such as the application of pipe rupture data by selecting an assessed range of failure rates below the composite nuclear/non-nuclear experience shown in Table III-1. Nuclear piping may be less susceptible to failure, but justification for the improvement is not developed in Appendix III. This example can be extended to any instance in which the WASH-1400 assessed ranges fall below best fit plots for composite nuclear/non-nuclear failure rate data.

Some of the data sources listed in Appendix III may be marginally applicable to nuclear systems. However, it is believed that the method of using this data in WASH-1400, a log-normal distribution with sampling from an assessed range, allows for use of this data.

As a final conclusion on data application, it is noted that major orders-of-magnitude of errors in selection of assessed ranges were not found. However, as noted in the detailed review, instances were identified where the assessed range was considered over-optimistic or over-conservative. The impact of these arguable range selections on accident probabilities should be evaluated on a selective, worst case, basis. While other applicable data sources can be found for specific failures, no major applicable compilation appears to be omitted. To obtain other data sources would be desirable but not necessary except in those instances where the data listed in Table III-1 are scattered or of such small sample size as to make the log-normal/assessed range approach technically weak.

(2) Have common mode failures been properly considered and accounted for?

This question cannot be conclusively answered from reviewing Appendix III and Appendix IV. Much of the specific treatment of common mode failures is discussed in other appendices. A major criticism of Appendix IV is the lack of a clearly defined, systematic approach to identifying and evaluating common mode failures. While there is an extensive listing of common mode failures in Appendix IV discussions, no quantitative evaluation of all listed failures is given.

(3) Has appropriate attention been given to consideration of design adequacy, including environmental effects, aging, etc?

Appendix X provides extensive engineering review of selected equipment and structures in the BWR and PWR plants used as reference designs for this study. These reviews evaluate both design calculations and actual installation, pointing out deficiencies in engineering design and construction which may be a source of a single failure or common mode failure. This information is then used in other appendices to modify basic failure rates and to assign frequencies to common mode failures. Thus, design adequacy has been addressed in considerable detail with respect to the two plants involved.

However, a generic treatment was not provided for design adequacy in typical BWRs and PWRs. Appendix X does not contain a summary discussion of the probability that specific equipment will be designed properly. At best, the summary tables in Appendix X provide a qualitative indication of which selected systems or equipment may be marginally adequate. One must look throughout the appendices to evaluate the question of appropriate attention to design adequacy. It is recommended that the question of probability of adequate design be addressed in Appendix X.

(4) Are the component failure rate values properly computed from data used by WASH-1400?

The log-normal distribution used in developing assessed ranges of failure rates appears reasonable. It is doubtful that any other statistical treatment would change the results of the study significantly. While the log-normal distribution appears suitable, the computation or selection of failure rate ranges from available data is questionable in some instances. Specific examples of this point may be found in the detailed review.

9. PWR Low Pressure Injection System Failure

General Review - A review was performed of the PWR low pressure injection system (LPIS) failure analysis as presented in Appendix II, Vol. 2 of WASH-1400. The analysis was found to be correctly done except for the consideration of operator error.

Figure II-54, page 299 of Section 5.6.3 (Appendix II, Vol. 2) of WASH-1400 illustrates the reduced fault tree for the LPIS system. This tree consists of single and double fault contributions. Under the double fault contribution, there is a subtree labelled "Failure in Pumps and Control of Cause Failure" (see Comment 3 under Miscellaneous Comments). This subtree consists of two identical branches for failures in "A" train components causing loss of the "A" train and failures in "B" train causing loss of the "B" train. Each branch contains two operator errors, each of which disables one train: valve A01 (1864A) closed or valve A03 closed for train "A", and valve B02 or valve B03 closed for train "B". This operator error probability is apparently assigned a value of 1×10^{-3} in all cases, since using this value to compute the double failure contribution produces the same value as the double failure contribution used in WASH-1400 (Figure II-57). Since it is expected that the controls of each set of valves (A01, B01 and A03, B03) are identical and close to each other, it does not appear realistic to assign the same operator error probability for closing the second valve in each set unless the first valve is closed inadvertently rather than deliberately. If the operator is motivated to deliberately close the first valve because he is confused, has incomplete or erroneous knowledge of the system, is following the wrong procedure, etc, then it would appear likely that he would deliberately close the second, since both valves in each set are identical and are installed in identical parallel lines and perform the same function. Appendix III of WASH-1400 states on page 130 that an error rate of 1.0 is assigned for the event "If an operator fails to operate correctly one of two closely coupled valves or switches in a procedural step, he also fails to correctly operate the other valve."

Assigning a conservative value of 1.0 for an operator error of closing the second valve in either set after he has closed the first (with a probability of 1×10^{-3} as was used in WASH-1400) increases the double failure contribution from 9.2×10^{-5} to 9.7×10^{-3}. Figure 19 is a reproduction of Figure II-57 in Appendix II, Vol. 2 of WASH-1400, modified to show the effect of this change. The modified values are shown in parentheses under the WASH-1400 values. As can be seen, this change in the double failure contribution increases the overall LPIS system unavailability from 4.2×10^{-3} to 1.4×10^{-2}, an increase of a factor of three. To assess the influence of this increase on the overall risk requires an analysis of accident sequences involving LPIS failure

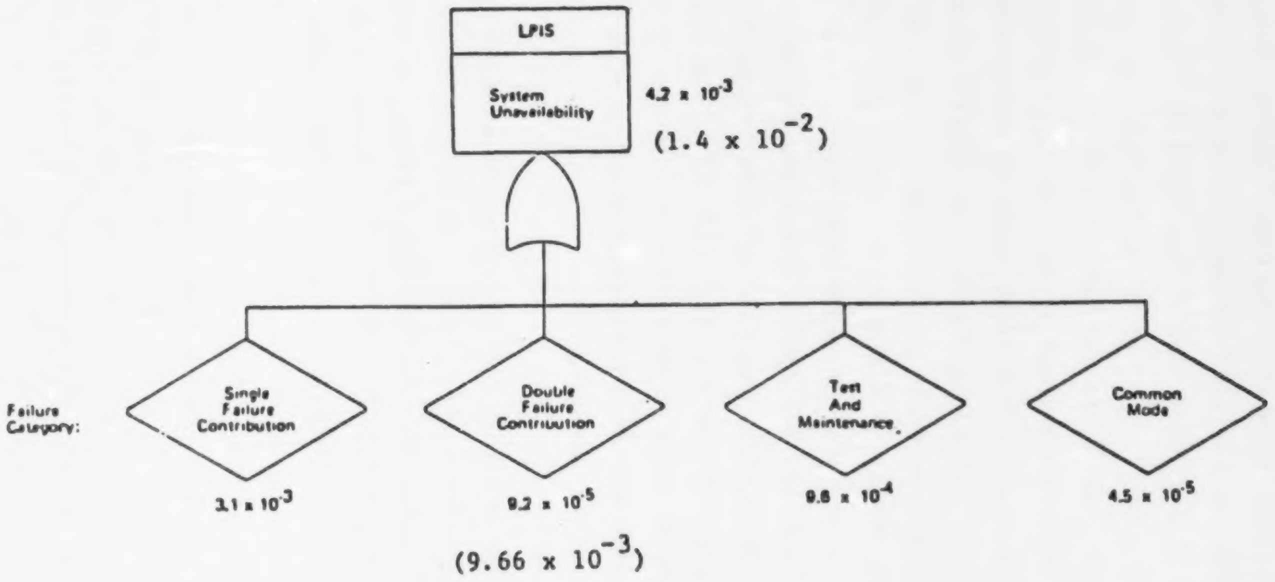

LPIS
System Unavailability
4.2 x 10-3
(1.4 x 10-2)
Failure Category:
Single Failure Contribution
3.1 x 10-3
Double Failure Contribution
9.2 x 10-5
(9.66 x 10-3)
Test And Maintenance
9.8 x 10-4
Common Mode
4.5 x 10-5

Figure 19 - Low Pressure Injection System Unavailability Contributions with Revised Numbers in Parentheses

(designated "D") as a dominant accident sequence for release categories 3, 4, 5 and 7 under large LOCA (the LPIS is required only for large LOCAs). However, the contribution from these sequences is less than 1 percent of the total from all accidents in each release category for Categories 3, 4 and 5 and about 3 percent for Category 7. Thus, a factor of three increase in the LPIS failure portion of the accident sequence would not have a significant effect on the overall risk result.

Omissions - No omissions requiring an in-depth review were found.

Trojan Comparison - A determination of the applicability of the Surry LPIS failure to the Trojan reactor resulted in the following:

(1) Pump capacities, injection pressures, and other pertinent parameters are the same for the Surry and Trojan plants. Piping layouts and system components arrangements are generally similar. Figure 20 shows flow diagrams for the two systems. The diagrams are similar except that the Trojan system pumps to four reactor coolant loops while Surry pumps to three. Indicated on Figure 20 by circled numbers are major single failure contributions causing total system failure. In all cases, these failures are valve failures. The Trojan system has three such contributions while the Surry system contains six. It would thus be expected, since single failures dominate the system unavailability (see Figure 20), that the Trojan system has a lower unavailability. However, in view of the small contribution to the overall risk from failure of the Surry LPIS, this difference would not be significant.

Conclusions - No errors or omissions of significance were found in the WASH-1400 analysis of the PWR low pressure injection system failure. Due to similarities in design and operation between the Surry and Trojan low pressure injection systems, the Surry analysis is largely applicable to Trojan. Because of fewer single failure possibilities, the availability of the Trojan LPIS should be somewhat higher.

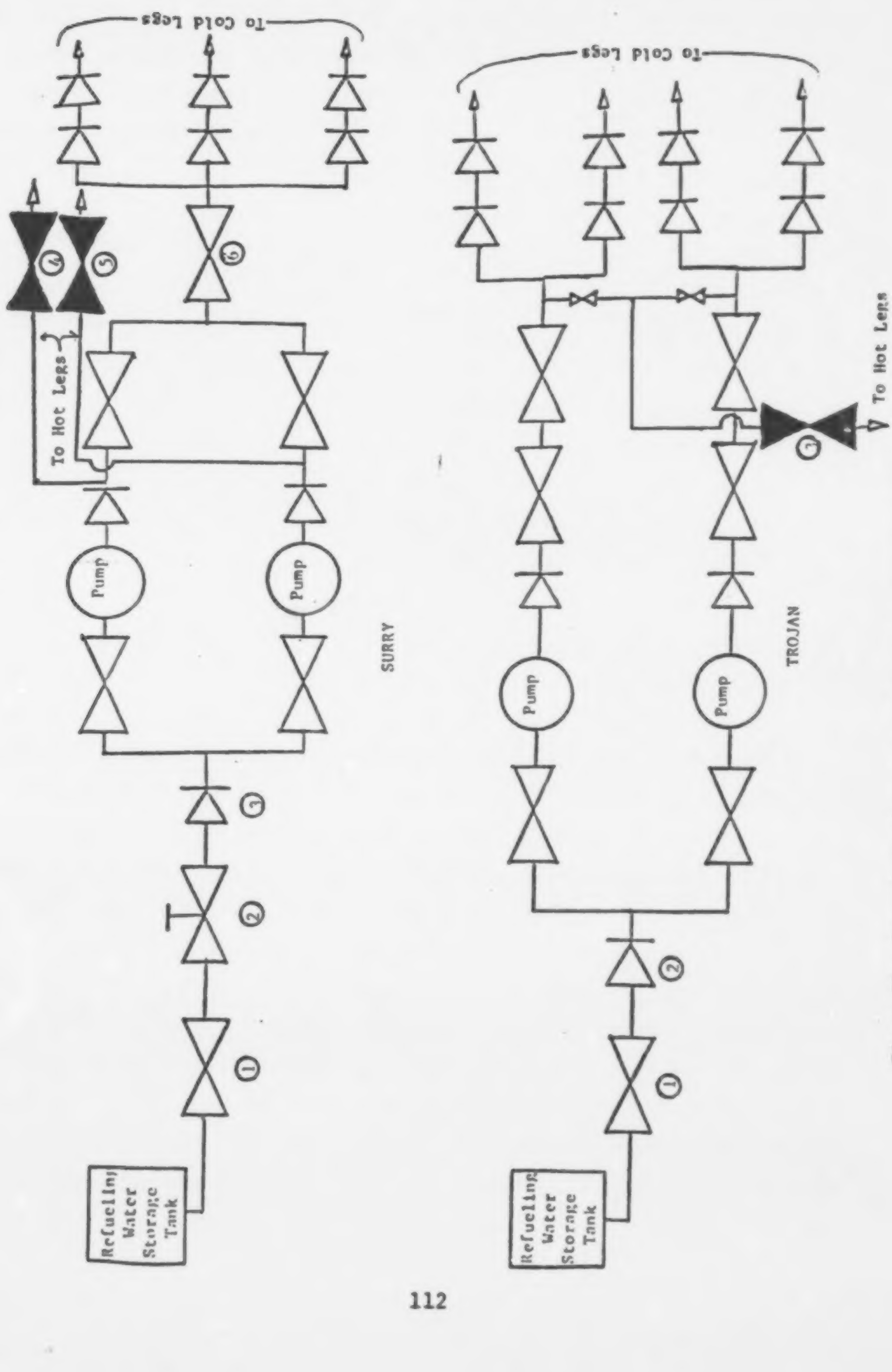

Refueling Water Storage Tank
①
②
③
Pump
Pump
To Hot Legs
④
⑤
⑥
To Cold Legs
SURRY
Refueling Water Storage Tank
①
②
Pump
Pump
③
To Hot Legs
To Cold Legs
TROJAN

Figure 20 - Comparison of Trojan and Surry Low Pressure Injection Systems

Miscellaneous Comments - In reviewing Section 5.6.3 of Appendix II, Vol. 2, the following comments were developed:

(1) Page 284 - The single failures here do not correspond to the single failures listed on the single failure subtree in Figure II-54. Further, neither considers the Refueling Water Storage Tank (RWST) drain plugged to be a single failure, as was done for the HPIS, where it was concluded to be not significant.

(2) Page 297, Figure II-53 - There appear to be inconsistencies between Figure II-53 and Figure II-65 (page 373). Figure II-53 shows two LPIS lines connecting to two HPIS lines outside the containment and then entering the containment and going to hot leg injection. Figure II-65 shows three LPIS lines connecting to three HPIS lines inside the containment.

(3) Page 299, Figure II-54 - The statement in the box at the top of the left hand subtree under "Double Cut Set" reads "(1) Failure in Pumps and Controls or Cause Failure." This statement is not clear and needs to be revised.

10. PWR Low Pressure Recirculation System (LPRS) Failure

General Review - A review was performed of the PWR LPRS failure as presented in Section 5.9, Appendix II, Vol. 2 of WASH-1400. The analysis was found to be correctly done except for a consideration of operator error. The operator action required to successfully initiate low pressure recirculation is not clearly specified in Section 5.9. On page 489, it is stated that the operator needs to open either or both of the valves in the pump suction lines to the containment sump (these valves are labelled V-24 and V-25 on Figure II-90, and 1860A (A05) and 1860B (B05) on Figure II-95) and also close the valve in the suction line to the Refueling Water Storage Tank (RWST). On page 495, the transition from low pressure injection to recirculation is defined as opening both V-24 and V-25. On page 498, it is stated that the RWST suction line valve need not be closed. For this analysis, it will be assumed that the operator only needs to open one valve; either 1860A or 1860B. The failure of the operator to perform this action is included as a common mode contribution to system failure in Section 5.9. The probability of operator failure for this action is assigned a value of 3.0×10^{-3}. Since the action required to open either valve is identical, the two acts are assumed to be completely coupled; ie, the probability of failing to open the second valve is one, given that the operator fails to open the first. Section 5.9 also assumes that operator failure probability to realign the LPRS injection to the hot legs from the cold legs after 24 hours is also 3×10^{-3}. The total common mode failure for continued LPRS availability, consisting only of these operator errors, is thus computed to be 6.0×10^{-3}.

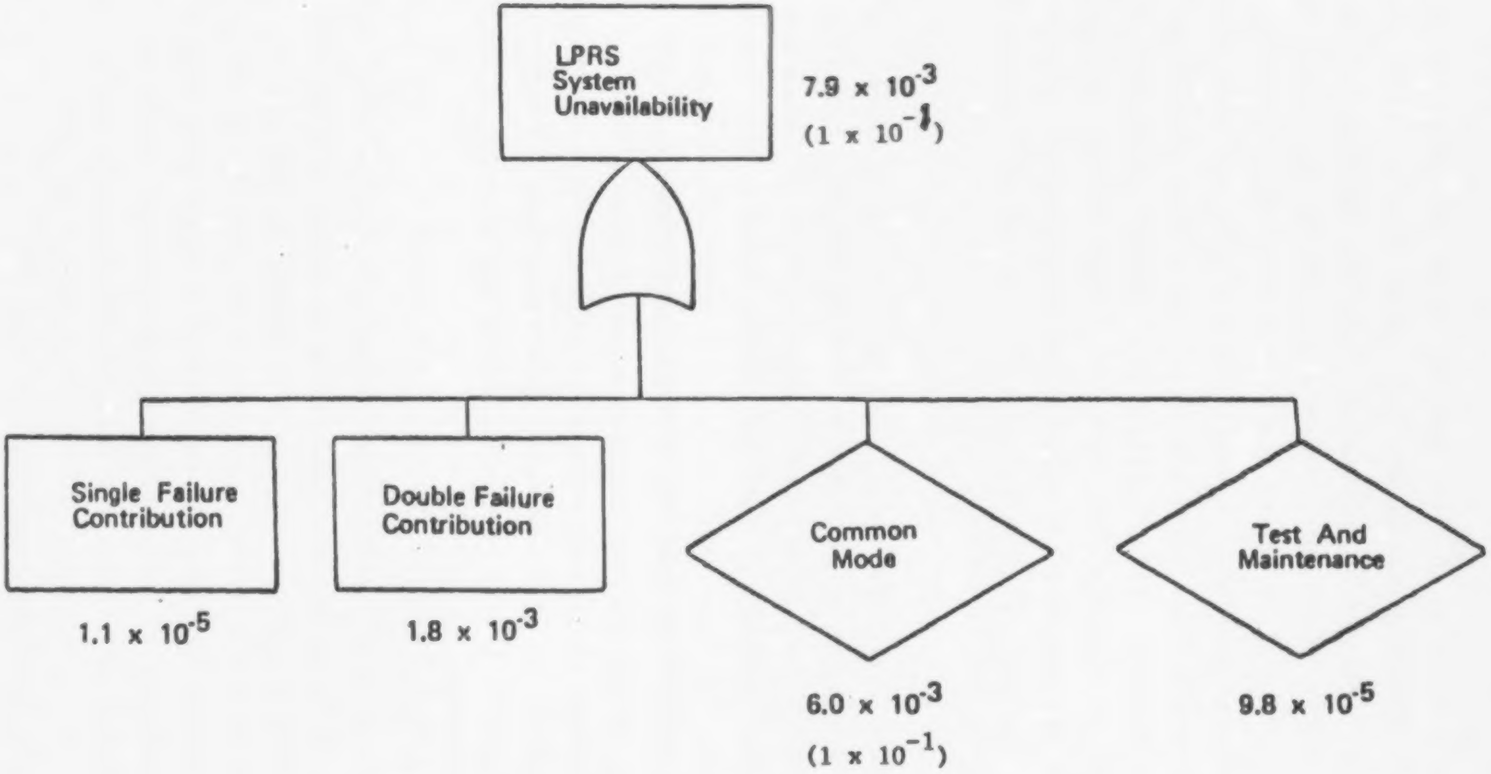
The probability of 3×10^{-3} assigned to the failure of the operator to open at least one suction line valve to the containment sump appears to be too low. The basis for the value is not clear. Appendix III, Section 6.1.2 of WASH-1400, discusses "Human Performance Data", and describes generally how human failures and errors were quantified and applied to the computations (in Appendix II) of system unavailability. This section includes a sample calculation of the probability that the operator fails to open either valve 1860A or 1860B due to (a) no action taken and (b) incorrect action taken. For (a), a negligible value was derived, and for (b) a value of 10^{-2} was selected. From this discussion, it appears that a value of 10^{-2} should have been used as the probability that the operator fails to open either of the valves, rather than 3×10^{-3}. In addition, it appears that 10^{-2} is also too low based on other considerations. The "basic error rate" for "Operator fails to act correctly after the first 30 minutes in an extreme stress condition" is given as 10^{-1} in Table III-13, page 131 of Appendix III. According to the text in Section 5.9, this basic rate was altered to account for other factors. However, the factors considered to justify the reduction to 10^{-2} are not delineated; a statement is merely made:

"The basic error rate of 10^{-1} was assessed to be too large for this type of action (opening containment sump valves for the LPRS), and 10^{-2} was accordingly selected as the nearest order of magnitude estimate." An apparent conflict with this assessment exists on page 135 of the same section. Here it is stated that: "The basic error rates in Table III-13 were modified by assigned (sic) higher rates to situations where the arrangement and labelling of controls to be manipulated were potentially confusing. For example, motor operated valves MOV-1860A and MOV-1860B..." The discussion goes on to describe how the operator could confuse the controls for these valves with others. This discussion seems to argue that the basic error rate of 10^{-1} is even too low.

It is recognized that many factors must be considered in order to determine with confidence the human error rate for an activity under a given set of circumstances. Such a determination is beyond the scope of this effort. However, it does not appear, based on the information presented, that the value of 3×10^{-3} used in WASH-1400, Appendix II (Vol. 2) for failure to open valves 1860A and/or 1860B is justified, and, further, it appears that a higher value is more appropriate. In order to determine the effect of this parameter on the overall risk assessment, a value of 10^{-1} was used. This value does not appear unreasonable based on information contained in Appendix III of WASH-1400 as discussed above.

The increase from 3×10^{-3} to 10^{-1} for operator failure to open valves 1860A and/or 1860B changes the LPRS system unavailability from 7.9×10^{-3} to 1×10^{-1}. This change is illustrated in Figure 21, which is a revised version of Figure II-96 from Appendix II, Vol. 2, of WASH-1400. The revised numbers are shown in parentheses. Translating this increase to probabilities for the PWR release sequences (Table V-16, page 37 of Appendix V) yields the following changes (all median values):

(1) Category 3 increases from 5×10^{-6} to 7×10^{-6}



LPRS
System
Unavailability
7.9 x 10-3
(1 x 10-1)
Single Failure
Contribution
1.1 x 10-5
Double Failure
Contribution
1.8 x 10-3
Common
Mode
6.0 x 10-3
(1 x 10-1)
Test And
Maintenance
9.8 x 10-5

Figure 21 - Revised LPRS Contribution Pictorial Summary

(2) Category 5 increases from 1×10^{-6} to 2×10^{-6}

(3) Category 7 increases from 6×10^{-5} to 1.4×10^{-4}

To assess the significance of these release category probability increases to the general population, the revised numbers were compared with the corresponding numbers in Table VI-20 (page 71, Appendix VI). Since only Categories 1 through 4 produce acute fatalities (average) of greater than one per accident, the increases in Categories 5 and 7 are of little consequence. The increase in Category 3 is small and thus would not significantly increase the overall risk.

Two other apparent problems with the LPRS reduced fault tree used in WASH-1400 were found during the general review. These are as follows:

(1) Figure II-91 - It is not clear why "operator error, operator closes MOV 1890C" is not included as a single failure under "Single Failures Which Can Cause Insufficient LPRS Flow."

(2) Figure II-91 - A subtree to the top event "Pump B01 Fails to Start and Pump A01 Fails to Continue Running" includes three valve faults. These valve faults (B03 and B02) do not have anything to do with pump malfunction. These valve faults should be included in the suction line (B03) and discharge line (B02) faults which are parts of separate subtrees B03. However, neither of these valve faults appear under suction line failures or discharge line failures. They do appear under the top event "Pumps A01 and B01 Discontinue Running." Again, these valves do not have anything to do with the top event. It thus appears that these valve failures have been counted twice, and in neither case have they been included where they logically belong.

<ins>Omissions</ins> - This review resulted in the finding of an additional failure mode which was not considered. Appendix I of WASH-1400 (page 24)

states that only cold leg breaks (for large LOCAs) were considered since this location presents the most stringent demand on ECCS. Appendix II (Vol. 2) states (page 490) that "...although no written procedures were found to be available, it was assumed that at some time during the first day following a cold leg break the LPR system should be realigned to inject into the hot legs..." Failure to perform this activity was assumed to result in failure of the LPRS. The probability assigned to this operator failure mode was $3x10^{-3}$. (In view of the fact that no emergency procedures were apparently available to the operator, this failure probability seems much too low.) It is not clear whether the operator will be able to distinguish between a hot leg and a cold leg break. If a hot leg break occurs, and the operator, assuming a cold leg break, switches to hot leg circulation, an LPRS failure may ensue for the same reasons as stated on page 490, Appendix II, Vol. 2. It is not expected, however, that such a failure would contribute to the overall risks because of the time elapsed (24 hours) before the action is taken and the relatively insignificant risk associated with LPRS failure, as discussed under the general review.

Trojan Comparison - This review resulted in determining that the pump capacities, injection pressures, and other pertinent parameters are the same for the Surry and Trojan plants. Piping layouts and system components arrangements are similar. Two exceptions to this similarity should be noted:

> First, on page 498 of Section 5.9, it is stated that ruptures in either pump discharge line up to the intersection of the two lines will not cause system failure because the downcomer in the pressure vessel will have been filled and the system need only deliver 300 gpm to be successful. Since Trojan operates at a power level about 1/3 higher than Surry, it will require an LPRS flow of about 400 gpm to be successful. Depending on relative pressure drops, a line break as described above may disable the

Trojan system. However, since line ruptures are relatively unlikely, this difference, if it exists, would not be expected to make any significant difference in the overall risks.

Second, part of the procedure for switching from low pressure injection to recirculation for Trojan includes aligning the LPRS such that part of the flow is injected into the containment sprays. The entire switchover for the Trojan (see Reference 2, Section 6.3.2.2.2) thus takes some 20 steps (versus 5 for Surry). This would increase slightly the probability of operator error, but the difference should not be significant.

Conclusions - The errors found in the WASH-1400 analysis of the Surry LPRS appear to be insignificant. No omissions which have a significant effect on the overall risks of nuclear power were found in the WASH-1400 analysis of the Surry LPRS. In addition, the analysis of the Surry LPRS should apply, with only minor modifications, to the Trojan LPRS.

Miscellaneous Comments -

(1) Appendix II, Vol. 2, page 490 - It appears that valve V10 needs to be closed to ensure hot leg injection. This is not listed in the procedure.

B. CONSEQUENCE AREAS

The analysis of the four consequence areas selected for review are presented here. For the core heatup and containment pressurization analysis contained in WASH-1400, a review was made of the assumptions and methods used. In addition, independent calculations using different techniques were performed and compared to the WASH-1400 results. For the PWR containment failure pressure assessment given in WASH-1400, a review was completed and a revised failure pressure was proposed based on information contained in WASH-1400 and the Surry Safety Analysis Report. For the fourth area, tritium release consequences following a major accident, a calculation was completed to show the biological effects of tritium release from the containment. An analysis of tritium release was not included in WASH-1400.

1. Parametric Studies of Core Meltdown (PWR)

This section presents a review of the PWR Core Meltdown analysis developed in Appendix VIII of WASH-1400. In addition, an independent analysis of selected core meltdown sequences is presented. The purpose of this section is to determine the applicability of the WASH-1400 core meltdown analysis as used in assessing the risks of nuclear power.

WASH-1400 Analysis - The computer code BOIL, described in Appendix VIII of WASH-1400, was written by Battelle Columbus Laboratories to calculate core heatup and meltdown during reactor accidents where water initially in the core is boiled away by the core fission product decay heat. BOIL was also written to calculate the core heatup for cases with very low bottom-flooding rates (<0.2 in/sec) and the dry heatup of the core in the absence of water. The code performs energy balances on the fuel rods and flow channel fluid to predict the water temperature and boiloff rate and the fuel rod temperature. Three different models are used to predict the heat transfer and molten fuel transport within molten fuel regions.

The BOIL code uses a simplified approach to solve a complex problem. The BOIL code solution is, however, consistent with the state-of-the-art, and the simplifying assumptions appear reasonable.

Independent Core Meltdown Analysis - In attempting to select a computer code to perform an independent core meltdown analysis, none could be found which could be directly used to calculate the effects of fuel slumping after the core begins to melt. Most of the codes currently being used for reactor thermal-hydraulics calculations are not intended for predicting fuel pin cladding temperatures above the 2200°F permitted by the ECCS acceptance criteria(36). It was decided to make the required modifications to the RELAP4 computer code(37) to permit a dry core heatup analysis until the hottest region in the core begins to melt. The core was assumed to be in a dry steam environment, but steam boiloff from the lower plenum due to heat transfer from hot reactor vessel walls was permitted. This case is representative of a LOCA with failure of all ECCS, and is recognized to be conservative because the passive accumulator systems will almost certainly provide some core cooling for some time after blowdown is complete.

The RELAP4 model used in the analysis (Figure 22) consisted of five fluid volumes: One each for the upper and lower plenums and three volumes in the core. Sixteen core heat slabs were defined, and two heat slabs transferred heat to the lower plenum. A hot core region representing the hottest 10 percent of the core fuel rods was also modeled by eight heat slabs connected to core fluid volumes in the same manner as the average core. The reactor primary system and fuel rod dimensions were obtained from the Surry SAR(3). The core axial and radial peaking factors were obtained from Battelle(53). The average fuel pins were represented by eight core heat slabs, two each of which connect to the fluid volumes at core inlet and outlet, and four to the central core fluid volume.

The heat transfer from the pressure vessel walls to the lower plenum water was modeled using two heat slabs representing the lower head and cylindrical walls up to the core inlet level.

The initial peak fuel rod temperature at the start of core heatup after blowdown was assumed to be 1200°F, and the reactor vessel wall temperature was assumed to be 571°F. The core decay heat was assumed to follow the ANS standard(55). The core heatup was assumed to begin 30 seconds after the start of blowdown. Runs were made with and without

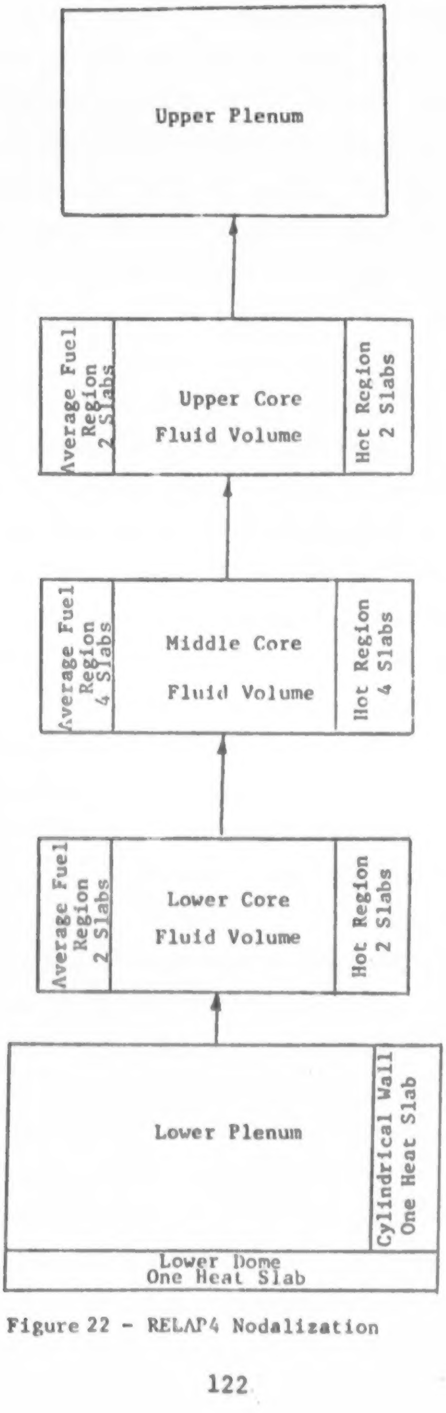

Upper Plenum
Average Fuel Region 2 Slabs
Upper Core Fluid Volume
Hot Region 2 Slabs
Average Fuel Region 4 Slabs
Middle Core Fluid Volume
Hot Region 4 Slabs
Average Fuel Region 2 Slabs
Lower Core Fluid Volume
Hot Region 2 Slabs
Lower Plenum
Cylindrical Wall One Heat Slab
Lower Dome One Heat Slab

Figure 22 - RELAP4 Nodalization

reducing the decay heat to account for fission product losses. One run was also made which assumed no cladding metal-water reaction.

The minimum time required to the start of a PWR core melting was assumed in WASH-1400 (see Appendix I, page 121) to be 16 minutes. The RELAP4 model described above predicts that core melting begins at 4.5 minutes if fission product release is neglected, and at 5.2 minutes when the BOIL code fission product release equations are programmed into RELAP4. If metal-water reaction is neglected, the core melting time is extended to 17 minutes. The case with no metal-water reaction corresponds to a case where no liquid exists in the lower plenum after blowdown, as was apparently assumed in the BOIL code dry heatup calculations. Thus, although boiloff of the residual water in the lower plenum does not provide significant cooling, it does provide enough steam for the metal-water reaction to greatly accelerate the heatup rate and, therefore, should be included in core heatup and meltdown analysis. It is not realistic to assume that no water will be left in the lower plenum following blowdown, especially if ECC accumulators are assumed to function.

The core heatup analysis in WASH-1400 assumes that the core is either initially covered by water, covered by water up to the 6-foot elevation, or completely dry. The first case corresponds to failure of the core heat removal system after the ECC systems have functioned successfully; the second apparently represents partially successful ECCS; and the last represents total failure of all ECCS.

The assumption of a dry heatup is conservative because the passive accumulator systems can be realistically assumed to provide core cooling for some time after the accident. A core reflood rate analysis was, therefore, performed as part of this review to determine the reflood rate history to be expected for a case with accumulator injection only. The analysis also predicts the core liquid level which can be expected at the beginning of core boiloff. The analysis must be considered as scoping in nature because of the unknown end-of-bypass time and the estimations used for accumulator injection rate and loop resistances. These values were selected on the basis of previous experience with similar calculations. One accumulator was assumed to inject into the primary system. The core carryover rate fraction was assumed to be constant at 0.8, based on PWR-FLECHT results(58) after the core water level reached one foot,i.e.,80 percent of the water predicted to enter the core was assumed to be carried out of the core to the containment. Figure 23 shows the core reflood rate and the core liquid level as a function of time after blowdown is initiated. The core reflood began at 20 seconds and continued until 200 seconds. The core liquid level at the end of the reflood was calculated to be 3.3 feet. Detailed calculations of the core temperature response during this time period would require computer code modifications which are beyond the scope of this study. However, examination of PWR-FLECHT(58) data shows that for an average reflood rate of 1 inch/sec, the core will be quenched to a level of less than 6 feet at 200 seconds. Most of the core will, therefore, be at temperatures greater than 900°F at the beginning of core boiloff. This is different from either the totally dry or totally quenched assumptions in WASH-1400.

The cooling effect of the short term reflood and the resulting 3-foot water level in the core can only be estimated. A reasonable estimate would delay the rapid core heatup calculated for an initially dry core by an estimated 5 minutes. The estimate assumes a 2-minute delay due to reflood and a 3-minute delay due to boiloff of the 3 feet of water. The 5-minute estimate could be increased due to level swell in the core or decreased because of increased metal-water reaction resulting from the higher steam supply. In either case, it appears that the time required for initiation of core melt during a LOCA accompanied by a loss of electrical power will be greater than the 4.5 minutes calculated herein for the no fission product release case, but less than the 16 minutes reported in WASH-1400.

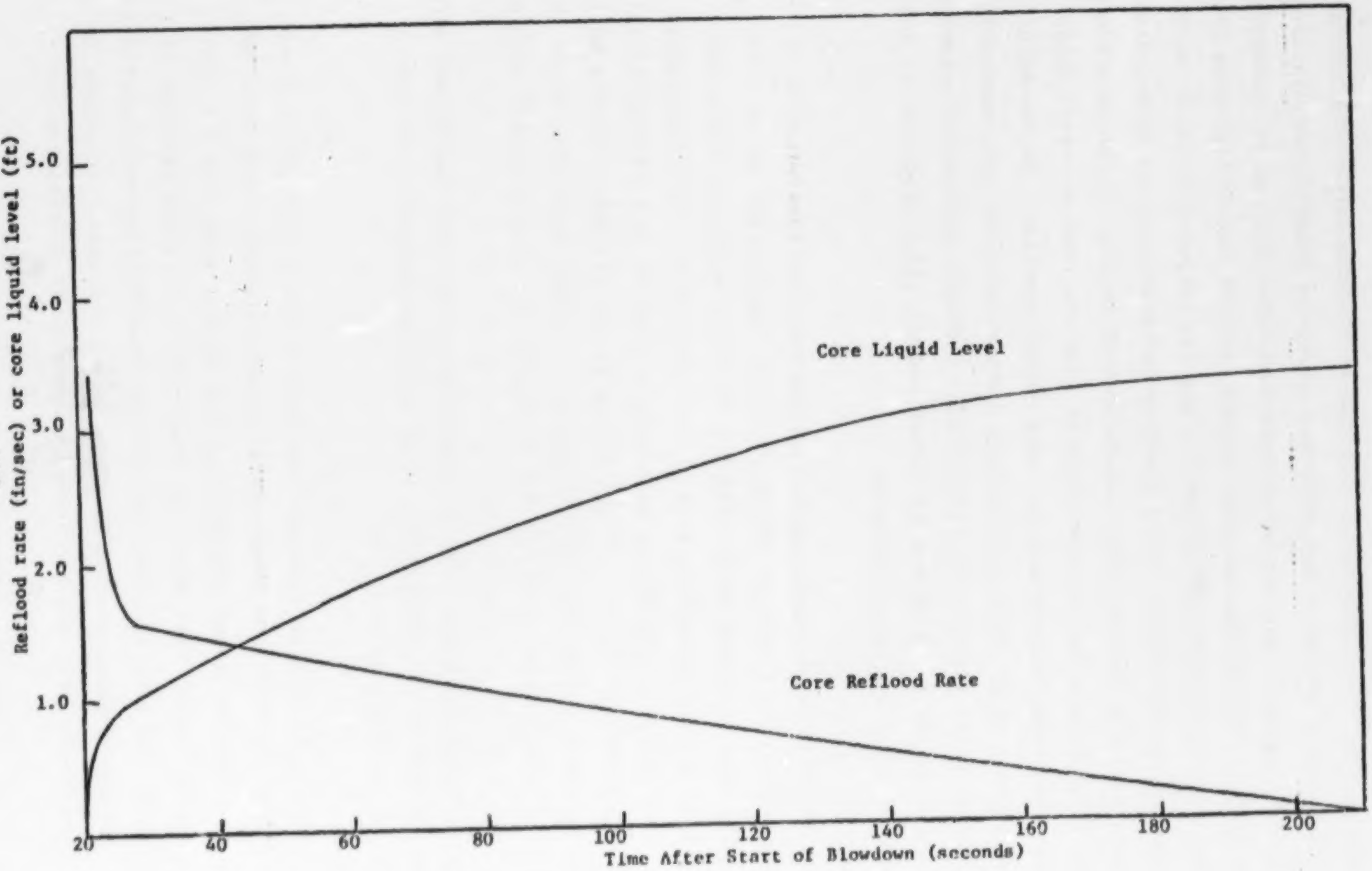


Figure 23 - Core Liquid Level and Core Reflood Rate as a Function of Time after Blowdown

Conclusions - An independent core heatup analysis for a LOCA accompanied by failure of all ECCS was performed using the RELAP4 computer code. The calculations were performed only until fuel melting was predicted to begin. The RELAP4 analysis showed the time to the beginning of fuel melt to be between 4.5 and 5.2 minutes compared to the 16 minutes reported in WASH-1400. The difference is apparently due to metal-water reaction. The RELAP4 model calculated a steam flow to the core from the lower plenum due to heat transfer from the reactor vessel walls thus resulting in a substantial metal-water reaction. In the BOIL code, there is no steam flow during a dry core heatup and, therefore, no metal-water reaction is calculated. A RELAP4 calculation assuming no metal-water reaction was performed and the calculated time to initial fuel melting was 17 minutes.

The impact of these observed differences upon the final results of WASH-1400 is difficult to assess directly. A decrease in the time to initiate core melt would slightly increase radiation levels and decrease evacuation times, both tending to increase the consequences of the accident. Additional analysis is required to determine if this is a significant effect. If the effect is significant, the core heatup case assuming accumulator injection needs further assessment since the calculations presented herein are different than the WASH-1400 results.

Miscellaneous Comments - Specific comments on the core heatup and BOIL code equations and assumptions as described in Appendix A of Appendix VIII are as follows:

(1) Page A-3 - The fission-product-release fraction is given by equation (A-2) for rod temperatures between 1500 and 2000°F and by equation (A-3) for rod temperatures greater than 2000°F. The report states that the value from one or the other equation is used. If this is true, the release fraction from both equations should be equal at 2000°F. This is not the case. Equations (A-2) and (A-3) could not be checked because their sources are not

referenced. In addition, the 1500°F initial cladding rupture temperature is inconsistent with cladding rupture temperatures cited in Appendix VII. Rod (cladding) rupture is assumed to begin at 1200-1400°F on page C-2 of Appendix VII, and a lower limit of 1400°F is stated on page 3.

(2) Page A-3 - The Baker-Just$^{(56)}$ rate law is assumed to be applicable until the fuel is completely melted. The applicability of the Baker-Just rate law has not been verified above the melting temperature of Zircaloy.

(3) Page A-4 - The radiation interchange factor given by equation (A-5) assumes the radiating and receiving surfaces to be parallel planes. Since this equation is used for calculation of radiation exchange between the fuel rods and the structure above or water below the core, the interchange factor should assume intersecting perpendicular planes.

(4) Page A-5 - Equation (A-6) assumes a "common gas" with a Prandtl number of 0.78$^{(35)}$. High temperature steam Prandtl numbers approach 0.88.

(5) Page A-5 - Equation (A-7) assumes a constant water temperature and a constant boiling heat transfer coefficient. These assumptions need to be justified since the system pressure and fuel rod heat flux are changing with time.

(6) Page A-6 - The Q_{melt} in equation (A-9) should be Q_{quench}.

(7) Page A-7 - Integration of equation (A-10) to get equation (A-10a) assumes $(T_{RT} - T)$ in last term on right-hand side of equation (A-10) remains constant during the time step. Since that term is variable, justification for this assumption should be provided.

(8) Page A-8 - The dry heatup model assumes steam and fission product flow from the core is due only to volumetric expansion of the gases. This is a good estimate if there is water in the lower plenum at a level high enough to seal the core inlet. If the core inlet is not water sealed, natural circulation through the core could transport the steam and fission products to the containment at a much higher rate than will volumetric expansion.

(9) Page A-11 - It is stated that the BOIL code does not include an entrainment model. This is not entirely true because, for the low or zero flooding rate cases considered, the mixture level model in BOIL accounts for all entrainment which will occur.

(10) Page A-12 - The core meltdown models should consider clad melting and slump occurring prior to fuel melting. The major effects would be earlier flow channel blockage for meltdown models A&B and reduced fuel mass and metal-water reaction for model C.

(11) Page A-14 - A possible fourth meltdown model should consider convection heat flux within the molten pool in all three directions. This would be consistent with the data of Hesson referenced in WASH-1400.

(12) Page A-18 - The steam generation rates given in Figure A-2 are wrong. The steam generation rate from decay heat in a full covered core one hour after shutdown is approximately 1900 lb/min.

2. Containment Response - Failure Pressure (PWR)

This section reviews sub-Appendix E of Appendix VIII of WASH-1400, entitled "Containment Failure Modes Evaluation." The purpose of the review is to determine (a) if the containment failure pressure selected for the Surry reactor, and used in other parts of the study in consequence assessments, is appropriately selected and reasonably justified, and (b) if the containment failure pressure assumed for Surry applies to Trojan, a reactor more representative of the reactor type which the results of the study are applied.

Surry Nuclear Plant - The Surry nuclear containment building(3) is a vertical circular cylinder with a spherical dome. The inside radius of the cylinder is 63 feet. The walls of the cylinder are 4-1/2 feet thick and consist of reinforced concrete. The dome is also reinforced concrete and is 2-1/2 feet thick. The inside of the containment is lined with a 3/8-inch thick steel liner which serves as a membrane to distribute the load to the reinforced concrete structure.

The structure is designed to contain a pressure of 45 psig with a safety factor of 1.5 with respect to yield of the reinforcing steel. Assuming proper design, proper construction practices, and use of materials which meet specifications, a pressure of 1.5x45 = 67.5 psig will cause yielding of the reinforcing steel. No credit for supporting strength is given to the steel liner in the design since it is to serve as a membrane only. The ACI Code(40) calls for a yield capacity factor of 0.9 to provide for small, adverse variations in material strength, workmanship, control, etc. Thus, 90 percent of the actual yield strength of the reinforcing steel was used as the design basis.

The Paul E. Mast consultant report contained in Appendix E of WASH-1400 gives the pressure which will cause yielding of the reinforced steel to be 75 psig. To obtain this value, a nominal safety factor of 1.5/0.9 = 1.67,instead of the design safety factor of 1.5,was used. Thus, the 10 percent reduction in design strength as called for in the ACI Code is essentially cancelled.

The Mast report presents an analysis which shows that,after yielding, the cracks in the concrete will increase from 0.03 inches to approximately 0.5 inches. This progressive stage of cracking during yield is accompanied by essentially no increase in load carrying capacity of the structure. The report states that the liner integrity will become important at this point and that the large strain at the cracks will most likely cause the liner to tear, resulting in loss of containment integrity. The report also states that one should not be misled by the fact that no cracks appeared around the penetrations during the load test and that the penetrations appear to be the weak link as far as radial shear is concerned. The report concludes that the failure pressure will be 80 psig, 5 psi higher than the 75 psi calculated to cause yielding of the reinforcing steel.

A report by S. G. Sampath, et al, is also included in Appendix E to support the selection of the containment failure pressure. This report indicates that the failure mode will be somewhat different from that predicted in the Paul E. Mast report. This study predicts that the failure mode will be due to large strains during yielding, which will break up the concrete between the inside and outside reinforcing steel. Thus, it will then be impossible to transfer the load from the liner and inner reinforcing steel to the outer reinforcing members. In this mode of failure, the strength of the containment is entirely dependent upon the liner and the inner reinforcement. Therefore, this report gives credit to the liner for a portion of the supporting strength of the structure. The fact that the liner is in biaxial tension is also taken into account, which gives the liner an apparent strength increase of a factor of 1.155. This factor is based upon the distortional yield criterion. The pressure at yield is then calculated to be 63.6 psig. The pressure at which the liner and inner steel will be at ultimate strength is calculated to be 87.5 psig. Thus, for this failure mode, the lower bound is 63.6 psig and the upper bound is 87.5 psig. The report concludes that failure will be somewhere between these two values, and predicts failure at 75 psig. (It should be noted that a strength reduction factor, as suggested by the ACI Code was not used in calculating these limits.)

Appendix E concludes on the basis of the Mast and Sampath reports, on pages E-2 through E-4, that the PWR (Surry) containment "...can be expected to fail at 100 ± 15 psia (85 ± 15 psig)." The conclusion is based on using the lower bound of 75 psig calculated by Mast, plus 17 psig attributed to the yield strength of the liner - something Mast indicated shouldn't be depended upon, and which seems to be in conflict with the statements at the bottom of page E-2: "It (the steel liner) has a relatively low strength in comparison with that of the reinforced concrete and depends on the latter for support. No allowance is given to the strength of the liner in the design of the structure."

In the evaluation of the Sampath report in WASH-1400, it is concluded on page E-4 that the predicted failure pressure (75 psig) may be overly conservative, and that the expected threshold of failure is taken to be the "approximate mean" of the Sampath and Mast (plus 17 psi) values. The mean between these two values is 83.5 psig, somewhat less than the selected value of 85 psig.

Trojan Nuclear Plant - The Trojan containment building(2) is a vertical circular cylinder with a spherical dome. The inside diameter of the cylinder is 124 feet, the wall is 3-1/2 feet thick and is pre-stressed reinforced concrete. The dome is 2-1/2 feet thick and is also pre-stressed reinforced concrete. The inside of the containment is lined with a 1/4-inch thick steel plate which serves as a membrane. The reinforced concrete structure includes both vertical and hoop tendons. These tendons are jacked to a stress level of $0.8f'_s$ or 145,000 psi. The ACI Code(40) calls for a reduction in strength of both the reinforcing steel and the stressing tendons. The reduced strength is 54,000 psi and 182,000 psi for the reinforcing steel and the tendons, respectively.

The design is such that the structure will contain a pressure of 60 psig with a safety factor of 1.5 with respect to yielding. Thus, if construction practices and quality of materials are at least par, when the internal pressure is 90 psig (104.7 psia), a state of impending yielding will exist and failure is imminent.

The cylindrical part of the containment structure has about 7000 psi compressive stress in the hoop reinforcing steel due to pre-stress. Upon a 90 psig loading, the stress in the hoop reinforcing steel will increase to 37,000 psi tension - an increase of 44,000 psi. The stressing tendons must strain as much as the reinforcing steel and their elastic moduli are equal; therefore, the stress in the tendons will increase to 189,000 psi (145,000 + 44,000). At this stress level, the tendons will have either yielded or be near the beginning of yield. A substantial portion of any pressure increase past this point will have to be supported by the reinforcing steel. A pressure increase of one psig above the design pressure could cause as much as a 7000 psi stress rise in the reinforcing steel. Thus, failure is impending at any rise above 90 psig.

Since the concrete is pre-stressed, the cracking at design pressure will be small. However, additional pressure beyond this point will cause the cracks to grow rapidly until a "weak link" develops. Some possible weak links are as follows:

(1) the concrete may crack to the point where it will no longer support radial shear. At this point, failure at the base could occur or a penetration such as the equipment hatch may blow out,

(2) the liner may tear at a point of high strain concentration,

(3) a pre-stressing tendon may snap, releasing a tremendous amount of strain energy and causing failure due to rapid crack propagation.

It is difficult to say exactly what internal pressure one of these weak links would support. However, since both the reinforcing steel and the pre-stressing tendons are at yield when the internal pressure is 90 psig, a slight increase in pressure will cause a large strain increase in certain areas. These strain concentrations could lead to immediate failure. The failure of one tendon would precipitate the failure of other adjacent tendons when stressed at this high level. For example, when the stress in a hoop tendon is 182,000 psi, the stored strain energy is 2,440,000 foot-pounds. If this tendon were to snap, this amount of energy would have to be immediately absorbed in the surrounding structure. If most or all of this energy were to be concentrated at a particular location, failure would certainly occur. Thus, when the structure is pressurized to the point that the tendons begin to fail, the complete structure will possibly fail in a catastrophic manner.

Several differences exist between the Surry and Trojan containment structures. The Trojan structure is pre-stressed, reinforced concrete - the Surry structure is reinforced concrete but not pre-stressed. The concrete in the Trojan structure is initially in compression due to pre-stressing. Thus, the cracks in the concrete at yield are smaller than those in the Surry containment. The analyses for the modes of failure should be somewhat different for the two structures. Each tendon in the Trojan containment is made up of 180 (1/4-inch diameter) steel wires. The tendon system is a primary component of the load support system. If the tendons fail under design load, the entire structure could fail catastrophically.

The liner plate in the Trojan is 1/4-inch thick steel - in the Surry, it is 3/8-inch thick steel. This difference is probably not too important since the liner plate will not influence the failure mode to a great degree in either structure.

Conclusions - The assumed failure pressure used in WASH-1400 for PWR containments of the Surry design, 85 psig, does not appear to be sufficiently justified, and appears to be too high. This conclusion is based on the following:

- The use by Mast of a safety factor of 1.67 instead of 1.5 is not justified.

- The addition of 17 psi in Appendix E of WASH-1400 to the failure pressure calculated by Mast appears to be improper since Mast indicates it should not be allowed and credit for liner strength is not permitted in the design. Under accident loading, the liner will be in compression due to thermal expansion. Thus, at the point where the reinforcing steel is starting to yield, the liner will not be supporting any of the internal pressure load. The liner, at this point, not only contributes nothing to the pressure carrying capacity of the structure, but due to thermal expansion, is exerting even more load on the reinforced concrete structure.

- Both the Mast and Sampath analyses stress that discontinuities in design (such as penetrations), insufficient design details, and possible imperfections in fabrication were not accounted for in the analysis, and could be important. It appears that the failure pressure should be reduced in an attempt to account for these factors. For example, Sampath argues:

  "It should be emphasized that the conclusions reported here must be considered as tentative. The reason is that, due to severe constraints on both time and the lack of details, many areas received only a cursory examination or were not pursued at all." (Page E-22)

  "A number of different failure mechanisms were considered and, so far as possible, quantitative estimates made for each. The accuracy associated with the estimated failure pressures was limited by the lack of specific design details." (Page E-23)

> "The liner and the reinforcing near the inside face are only inches apart. This circumstance probably required very careful workmanship during construction to achieve proper consolidation of the concrete and its integrity is somewhat suspect. However, assuming that this section is properly constructed..." (Page E-24)
>
> "The structure has an excess of steel (in the neighborhood of 5 percent whereas it is usual to find 1-2 percent steel in typical well-designed concrete structures). Proper compaction of concrete under such conditions would be difficult to achieve." (Page E-29)
>
> In addition, the section in Appendix E which concludes that the failure pressure is 85 psig contains a discussion of the effect of penetrations, discontinuities and other factors. It is concluded on page E-4 that "Such considerations are difficult to quantify, at best, and cannot be evaluated in the absence of specific details on design, quality of workmanship, and results of nondestructive examinations. Other factors bearing on the failure potential of major penetrations include the manner in which they are anchored to the concrete and the reinforcing steel, the reinforcing of the liner around such penetrations, and the attachment of the liner to the penetrations. The significance of these factors will obviously vary with the details of each design." (Page E-4)

In view of the above considerations, it appears unjustified to use a containment failure pressure of 85 psig (100 psia) for the Surry containment building. Due to the complex nature of the containment structure, uncertainties in the design, fabrication and material quality, all that can be said with any degree of certainty is that the building will withstand a pressure to which it is designed and tested. (Acceptance testing is required$^{(41)}$ at 1.15 times design.) In addition, the building will probably hold 1.5 times design pressure, or 67.5 psig (82.2 psia) as discussed previously. In the absence of any destructive testing information, any pressure it will contain above this value is speculation. It is probable that the building will withstand pressures in excess of this value, but in view of the rapid stress buildup with pressure and other uncertainties discussed above, the unknown excess could be very small.

Regarding the applicability of the Surry containment failure pressure to the Trojan containment building, the design differences discussed above would render such application invalid without further supporting analysis. In the concluding discussion, it is stated on page E-1 that "To the extent possible, the extension of the results to other containments of similar design has been indicated." There is no indication of any such extension in the conclusions other than the results apply to a "reinforced-concrete containment building design for an internal pressure of 60 psia..." Containment buildings of the more representative Trojan PWR design are designed for 75 psia. Even though the Trojan and other 4-loop PWRs are designed for a higher containment pressure, the loss-of-coolant accident produces a higher pressure. The effect of these differences upon the time at which containment overpressure failure occurs is discussed in detail under item 3 of this section.

Miscellaneous Comments - During the review of sub-Appendix E of Appendix VIII in WASH-1400, the following comments were developed:

(1) Throughout sub-Appendix E, units of psi, psia, psid, and psig are intermixed, which tends to confuse the quantitative assessments of containment pressure.

(2) A pressure of 5 psi is added to the computed failure pressure in the Paul E. Mast section of sub-Appendix E. There does not appear to be any justification included for the increase.

3. Containment Response - Pressure History (PWR)

This section presents the results of a review of the WASH-1400 containment response analysis for the Surry reactor plus two independent calculations of PWR containment response.

WASH-1400 Analysis - Appendix VIII contains little discussion of the analysis methods and assumptions used to calculate the reported containment response. Discussions with BMI personnel(53) revealed that the containment pressure behavior calculations were made without the use of a computer, using simple energy and mass balances. Computer codes, such as the CONTEMPT-LT Code(39), are readily available and can be used for transients of the type investigated in WASH-1400.

The assumptions used for the WASH-1400 PWR containment pressure response analysis are discussed on pages A-27 through A-30 of Appendix VIII. The discussion includes the following major topics: Energy Sources, Containment Heat Sinks, Containment Atmosphere, and Containment Leakage.

The general approach used in WASH-1400 in treating the potential energy sources was to justify ignoring some sources by assuming some of them to be offset by some potential energy absorbing effects. For example, all of the decay heat is assumed to result in steam generation while the metal-water reaction energy is assumed to go entirely into heating up the reactor core. The additional water flow to the containment due to entrainment during core reflood is neglected. These assumptions may not affect the end results of the study, but more rigorous assumptions appear to be required in the absence of supporting information justifying the assumptions used.

A constant steam-condensing coefficient of 150 Btu/hr-ft^2-°F was assumed for all "cold" heat sinks within the containment. This assumption while not consistent with experimental data and state-of-the-art containment analysis(54), is not expected to greatly affect the long-term containment response of interest. The major heat sinks in the containment are concrete and the heat transfer to concrete will become conduction limited early in the transient. However, the analysis could have gained credibility through the use of state-of-the-art correlations for condensation coefficients rather than using a gross assumption.

The containment leakage assumptions in WASH-1400 appear reasonable compared to usual containment design values. The source and justification of the assumed leakage rates should, however, be provided in the report.

Independent Surry Containment Analysis - The CONTEMPT-LT(39) computer code was used to calculate the Surry containment pressure response for two LOCAs. The LOCAs chosen for analysis were: (1) a large cold leg break accompanied by loss of all electric power to the emergency safeguard systems, including containment sprays and emergency core cooling systems; and (2) a large cold leg break (pump discharge) accompanied by loss of containment sprays.

The CONTEMPT-LT model included simulation of containment dry well, the reactor primary system, and engineered safeguard systems. The model also included heat conduction to or from various structures within both the containment and the reactor primary system. The system dimensions and initial conditions were obtained from the Surry SAR(3). The heat transfer coefficients for heat transfer to or from containment and primary system structures were also obtained from the SAR. The primary coolant mass flow to the containment during blowdown was obtained from the SAR and the fluid enthalpy was defined by the primary system energy content prior to blowdown. The mass flow table in the SAR does not include the reflood phase of the accident. For this analysis, reflood was assumed to begin at 15 seconds and continue until an assumed core quench time of 600 seconds. The delivery rate to the core was assumed to be equivalent to a cold reflood rate of 6 inches/second for 4 seconds, 1.5 inches/second to 300 seconds, then decreasing linearly to zero at 600 seconds. The remainder of the ECCS delivery was assumed to spill to the containment sump. Eighty percent of the reflood water entering the core was assumed to be carried out the top of the core and through the steam generators, where it was heated to the secondary system temperature before flowing to the containment. After 600 seconds, the mass and energy flow from the reactor primary system is due to the core decay heat and conduction of residual heat from primary system metal. For the case where all electric power (including that to the ECCS) was assumed to fail, the core reflood was assumed to be as predicted in Section V B-1 of this report.

The Surry containment pressure response as predicted by the CONTEMPT-LT model for the assumed failure of the containment spray injection system (CSIS) and the containment spray recirculation system (CSRS) is shown in Figure 24. The corresponding curve from WASH-1400 is also shown in Figure 24. There is a significant difference between the two curves for the first ten minutes of the transient. There are two major reasons for the differences. The decrease in containment pressure predicted by CONTEMPT-LT is due to high heat transfer rates to cold heat sinks in the containment. The only apparent significant difference between the WASH-1400 and CONTEMPT-LT models is in the modeling of the containment walls and dome. The WASH-1400 model considered only the steel liner and the CONTEMPT-LT model considers the steel liner and the concrete walls as a composite slab with negligible contact resistance between the steel and the concrete. The pressure rise occurring between 3 and 10 minutes predicted by CONTEMPT-LT is due to the liquid carryover during core reflood. The WASH-1400 analysis neglects this effect. After 10 minutes, the reactor decay heat becomes the controlling factor in the containment pressure transient and, thus, the pressure transients from the two analyses are nearly equal. The WASH-1400 assumed failure pressure (100 psia) is shown along with a revised lower limit failure pressure of 82.2 psia as discussed previously in Section V B-2 of this

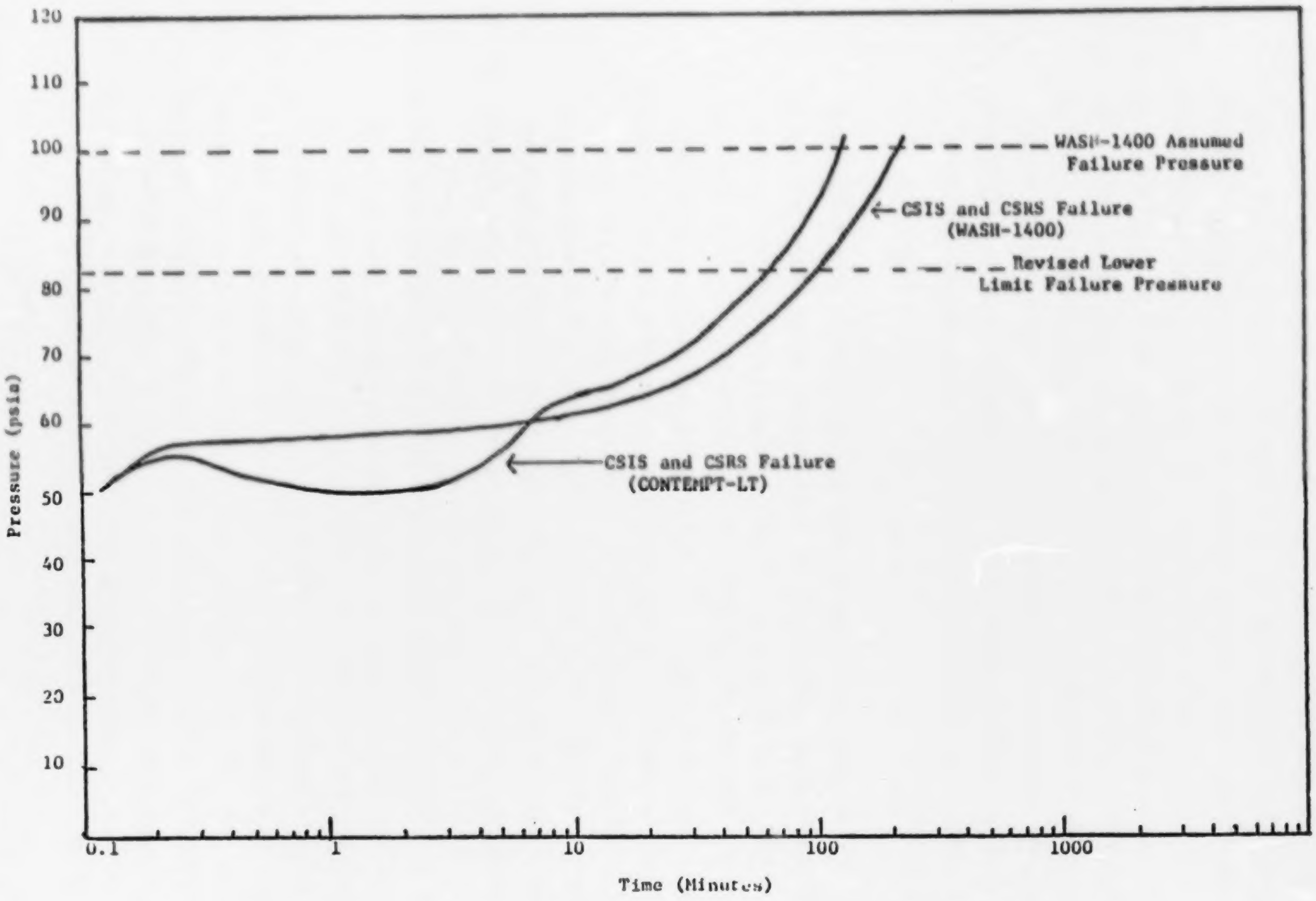


Figure 24 - Surry Containment Pressure During LOCA with CSIS and CSRS Failure

report. The CONTEMPT-LT analysis along with the revised failure pressure predicts containment overpressure failure at 63 minutes compared to the 230 minutes predicted in WASH-1400. The fact that the offsetting effects of containment heat loss and core reflood entrainment reduces the difference between long-term containment pressures predicted by the CONTEMPT-LT and WASH-1400 analyses is fortuitous and cannot be assumed to occur during all accident sequences.

Figure 25 compares the CONTEMPT-LT and WASH-1400 results for the LOCA accompanied by a complete loss of electric power. The pressure versus time curves are terminated when the reactor pressure vessel water has completely boiled away. (CONTEMPT-LT is incapable of calculating beyond this point.) Neither case predicts containment failure, but CONTEMPT-LT predicts a peak pressure of 81 psia which is nearly equal to the revised lower limit of 82.2 psia. If combustion of the hydrogen generated by metal-water reaction in the core is considered, the pressure transients become as shown in Figure 26. The WASH-1400 analysis exceeds the 82.2 psia failure pressure in 25 minutes and reaches 100 psia at 70 minutes. Heat transfer to structures within the containment delays the containment failure time predicted by the CONTEMPT-LT model until 174 minutes. The water in the reactor pressure vessel is completely boiled away before the pressure calculated by CONTEMPT-LT reaches the WASH-1400 failure pressure of 100 psia.

The containment failure mode and time at which the failure occurs determine the extent of the fission product release, the radiation level of the release, and the effectiveness of the population evacuation procedures. The CONTEMPT-LT analysis of the Surry containment and redefinition of containment failure pressure performed in this study (see Section V B-2) indicates significant differences from those in WASH-1400 for containment failure modes and times. For the two LOCA cases analyzed, one was found to result in a containment overpressurization failure at a much earlier time than predicted in WASH-1400 and the other

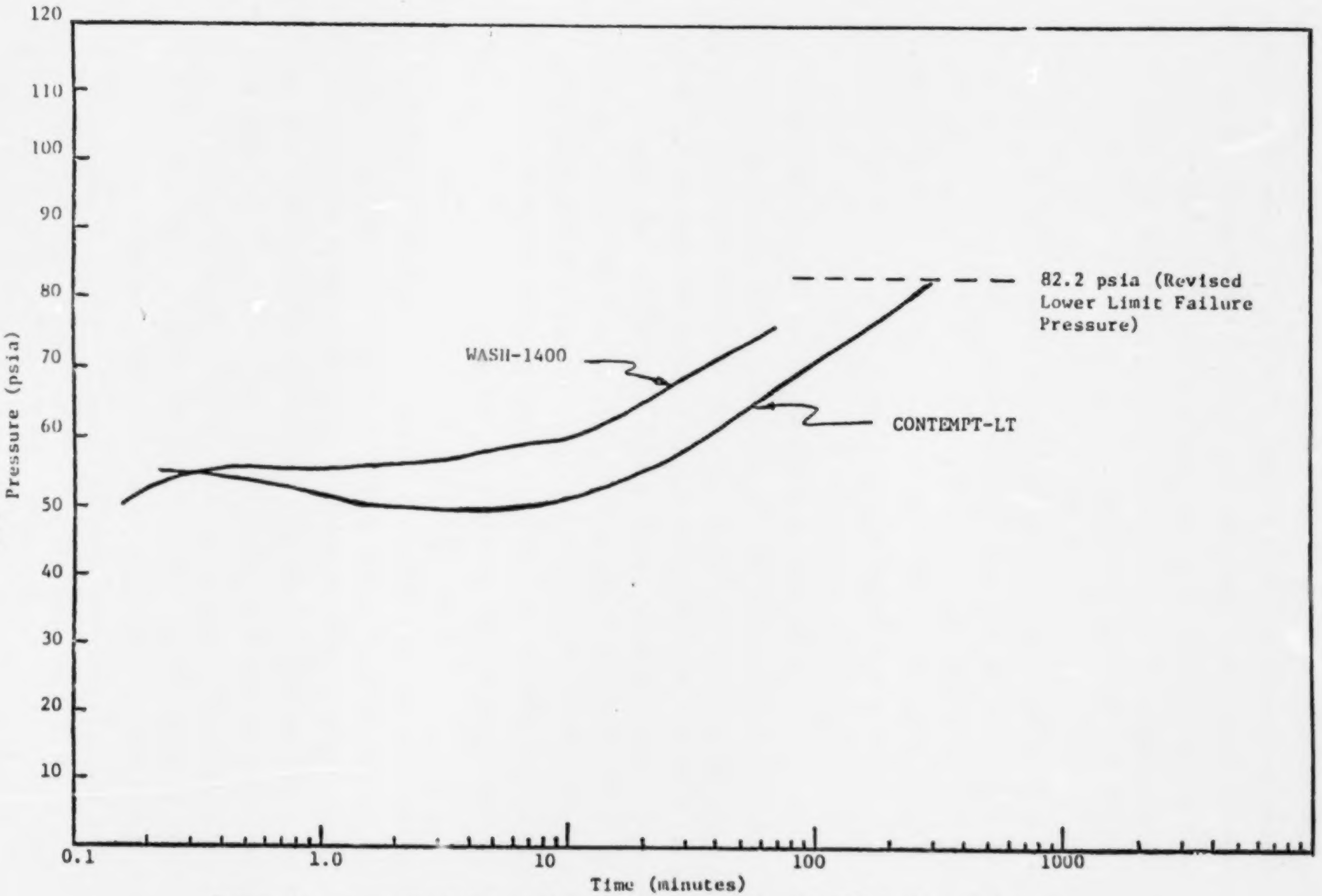


Figure 25 - Surry Containment Pressure During LOCA with EPS Failure and No H_2 Combustion

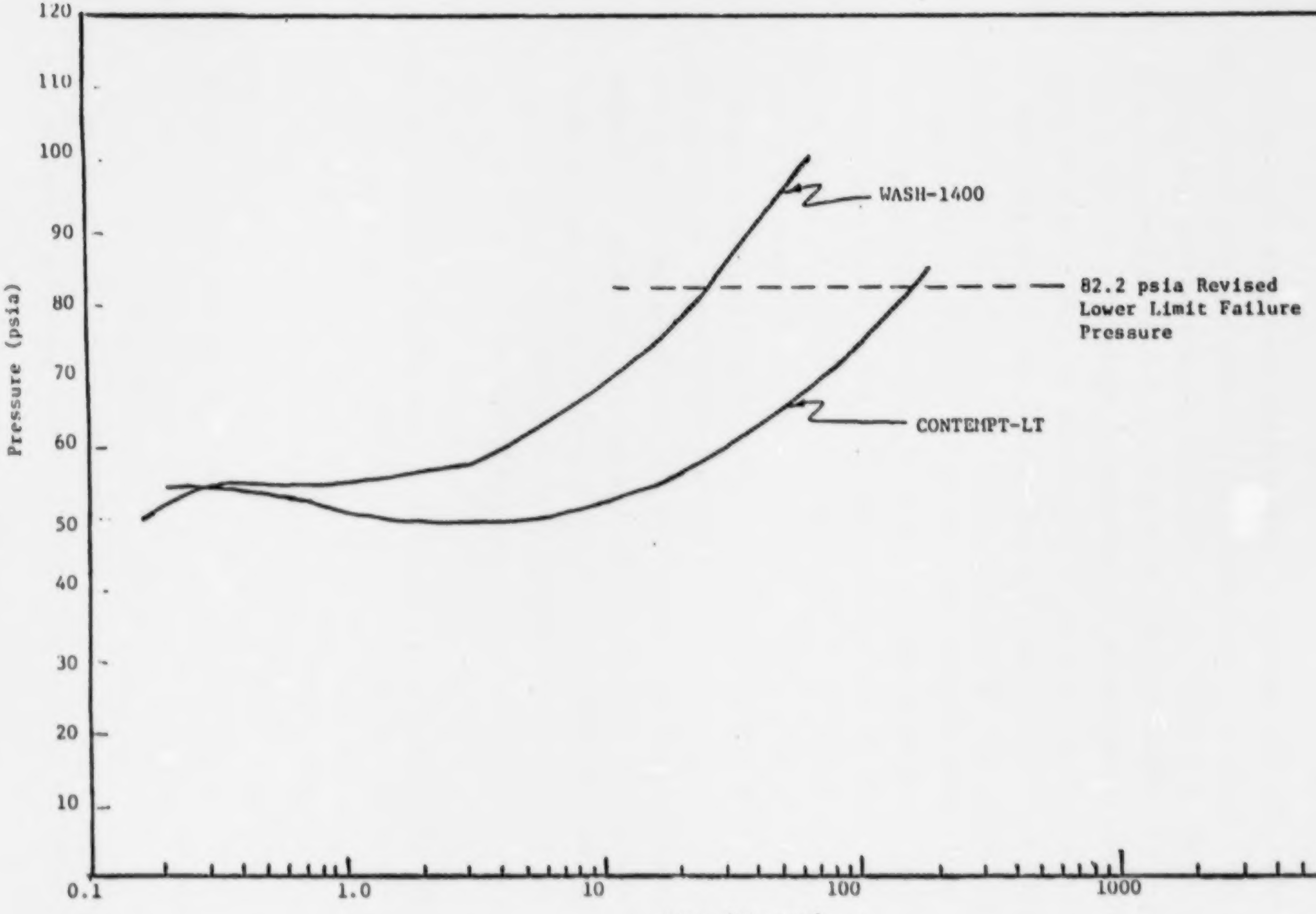


Figure 26 - Surry Containment Pressure During LOCA with EPS Failure and H_2 Combustion

case predicts that the containment overpressure failure predicted in WASH-1400 will not occur until a significantly later time.

An investigation was undertaken to determine the significance of the difference between the containment failure times assessed in WASH-1400 and the times calculated herein. The containment failure time will influence the radioactive source term available for release, the time available for evacuation of the population potentially available for exposure, and the probability of containment failure by melt-through as opposed to overpressure. The radioactive source term available for release will decrease as the containment failure time is extended due to (1) increased deposition and plateout of the radioactive material on internal containment surfaces, and (2) extended radioactive decay time. The change in the amount of fission product deposition and plateout in the containment is not a strong function of time as pointed out in Appendix VII (Appendix J) of WASH-1400. The reduction in the source term from radioactive decay for the times involved in the containment failure calculations is also not significant. An indication of the composite effect of decay time and deposition and plateout on the radioactive source term can be obtained from Appendix VI of WASH-1400. As shown in Table VI-2 of that appendix, an increase in time of release from 1.5 to 2.5 hours has an insignificant effect on the fraction of core fission product inventory released.

It was not possible, due to uncertainties associated with the application of the evacuation model used in WASH-1400 (see Miscellaneous Comments at the conclusion of this analysis), to determine quantitatively the effect of changes in evacuation times associated with the calculated containment failure times. However, a simplified analysis was done in an attempt to determine the sensitivity of the accident risks to evacuation times.

WASH-1400 predicts that containment overpressure failure at 100 psia will occur at 230 minutes and the revised 82.2 psia lower limit is reached at 105 minutes for a LOCA accompanied by loss of containment safeguards. Using the failure pressure of 82.2 psia recommended in this study and the CONTEMPT-LT analysis results in a containment overpressure failure at 63 minutes. The CONTEMPT-LT analysis predicted 100 psia at 140 minutes. One effect of this difference will be upon the population evacuation effectiveness. (It should be noted that the validity of the WASH-1400 evacuation model has been questioned by the Environmental Protection Agency(58).) The magnitude of this effect can be estimated by examining the evacuation model equation given on page 31 of WASH-1400 Appendix VI as

$$F_i = a+(1-a)\exp\,[-\lambda(t_i-TL+T_j)] \qquad (1)$$

where

F_i = Fraction of population remaining after evacuation
a = Fraction of population unaffected by evacuation
λ = Measure of the evacuation rate
t_i = Transport time to mesh point i
T_j = Time between awareness of impending core melt and leakage for accident type j
TL = Time lag associated with interpretation of data and issuance of warning to evacuate.

Assuming $a = 0.1$, $TL = 0.02$ day, $\lambda = 8.3$ days^{-1}, and a 20-mile radius to the population as in WASH-1400, and further assuming a 2.5 meter/second wind velocity, the above equation (1) becomes

$$F_i = 0.1+0.9\exp\,[-8.3(T_j+0.129)] \qquad (2)$$

Figure 27 is the plot of equation (2). If t_j is assumed to equal the time to containment overpressure failure, the WASH-1400 failure time of 230 minutes allows for all but 18 percent of the population to be evacuated. The 63 minute failure time predicted in this study results in failure to evacuate 31 percent of the population. Thus, the number of people exposed will be increased by 72 percent if the results of this study are used.

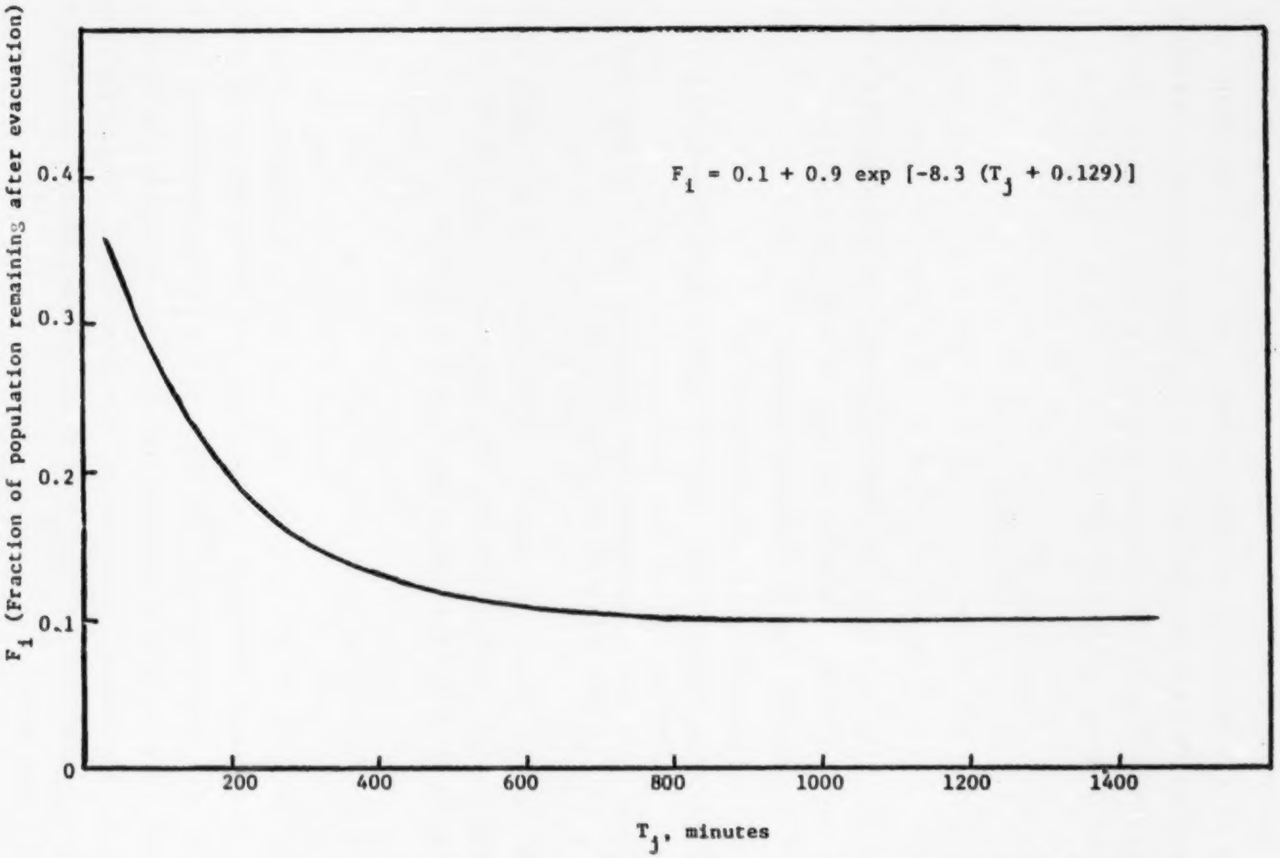


Figure 27 - Evacuation Effectiveness versus Containment Failure Time (T_j)

For the assumed LOCA accompanied by complete loss of electrical power, WASH-1400 predicted containment failure time is 70 minutes. This would result in 30 percent of the population being subject to exposure. The corresponding analysis in this study shows that containment overpressure failure and significant leakage will not occur until 174 minutes. For this containment failure time, the fraction of the population subject to exposure obtained from Figure 4 is 21 percent, a reduction of 9 percent from the WASH-1400 value. Neither of these changes is of major significance.

The effect of changing the probability of containment melt-through as opposed to containment failure by overpressure could not be assessed due to lack of information. If the containment failure is predicted to occur earlier for a particular accident sequence, then the probabilities of containment failure by melt-through could be reduced, and some accidents which were calculated to result in melt-through failure could change to overpressure failure and the risks would increase since overpressure failure presents a substantially greater hazard to the exposed population. It is stated on page 31, Appendix VIII of WASH-1400, "Since overpressure (failure) of the containment and containment melt-through can occur at approximately the same time, it is necessary to consider the competition between the two events." A reassessment of these two failure modes and their effect on the accident risks for the appropriate accident sequences appears to be needed based on the results of the containment overpressure failure times calculated herein.

Trojan Containment Analysis - To assess the assumed general applicability of the WASH-1400 results to all PWR power plants, a scoping analysis was performed on the Trojan Containment System. The purpose of the analysis was to determine which assumed loss of safeguards would result in containment overpressure failure during a large break LOCA. In modeling the Trojan containment with the CONTEMPT-LT computer program, only the containment volume and its associated heat absorbing structural components were considered. The mass and energy flow from the reactor primary system were supplied in table form from data provided in the Trojan SAR(3). The containment safeguards systems, when effective, were assumed to operate at the times and capacities specified in the SAR. No attempt was made to account for events such as the molten core dropping into a pool of water, hydrogen combustion, CO_2 formation as the core melts through the containment floor, or failure of the emergency core cooling systems. The LOCA chosen for this analysis was the double-ended pump suction guillotine break designated in the Trojan SAR as the design basis accident. The containment overpressure failure pressure was assumed to be 90 psig as discussed in Section V, B-2.

The results of the Trojan containment analysis are presented in Figure 28. Containment overpressure failure was found to occur at 50 minutes when all containment safeguards fail and at 190 minutes when the fan coolers and low pressure recirculation system are assumed to fail. These failure times bracket the 63-minute containment failure time predicted for the Surry LOCA with loss of containment safeguards.

Conclusions - An independent assessment of the Surry containment response analysis indicates that (1) for the case assuming CSIS and CSRS failure, the containment failure time will be reduced from the WASH-1400 time of 230 minutes to 63 minutes, and (2) for the case assuming complete loss of electrical power, the containment failure would occur about 100 minutes later than predicted in WASH-1400. Except for the possible effect of changing the probability distribution between containment failure by overpressure and failure by melt-through, which was not assessed, the effect of the different containment failure times does not appear significant.

The Trojan analysis indicated that the Trojan containment failure time is comparable to Surry. However, the differences in containment failure pressure and engineered safety systems for the two reactors may lead to a different assessment of risks. A detailed assessment for the

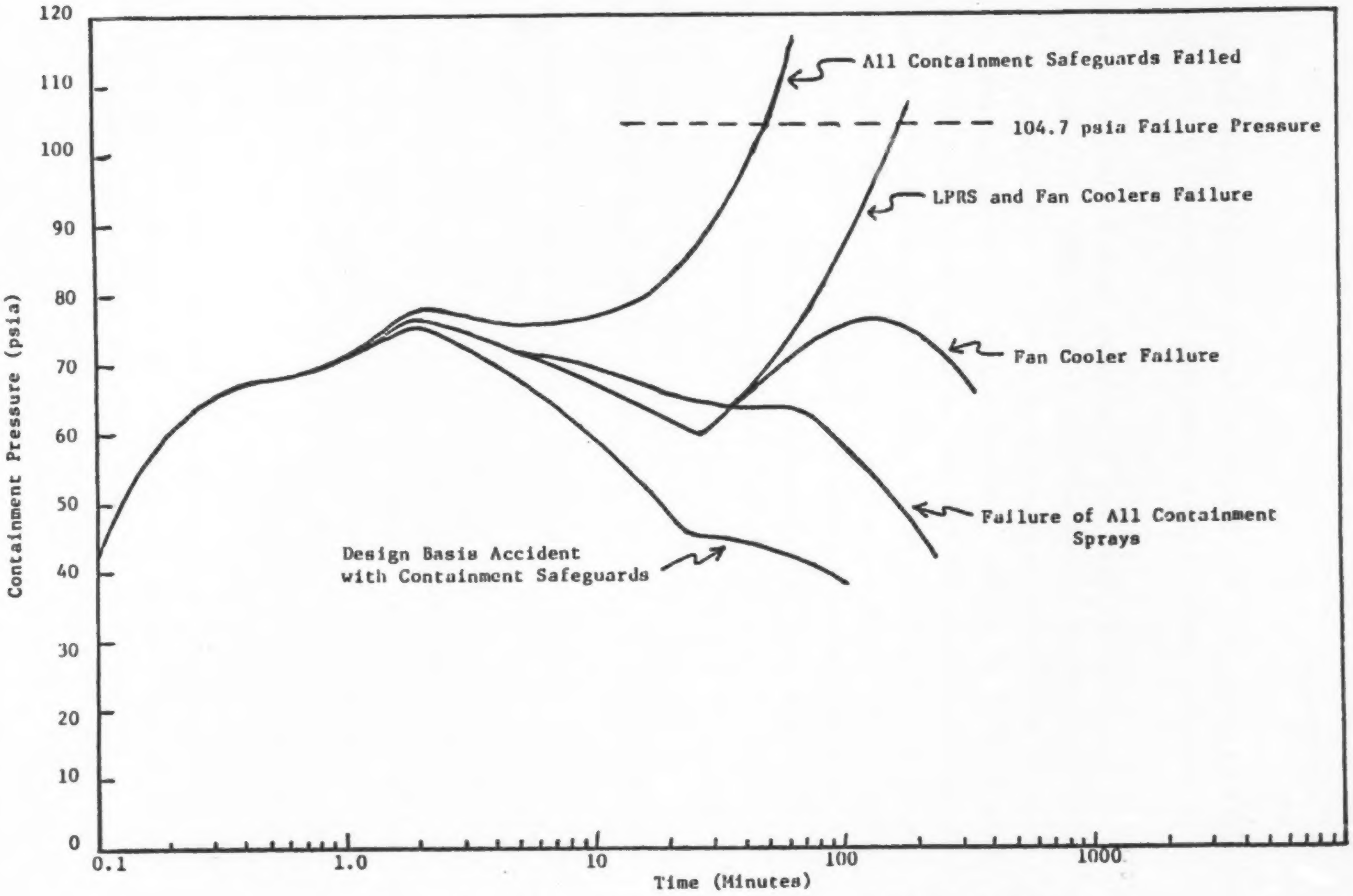


Figure 28 - Trojan Containment Pressure During LOCA with Assumed Containment Safeguards Failures

Trojan design would be required in order to establish the significance of these differences.

Miscellaneous Comments - The following comment was developed during the review of the containment failure assessment contained in WASH-1400:

(1) It is not entirely clear how containment failure times and evacuation times were computed and used in WASH-1400. The time of release of the fission products given in Table VI-2 (Appendix VI, page 9) does not appear to correspond to the calculated time of containment failure for PWR release Categories 2 and 3 which are dominated by accidents culminating in containment failure by overpressure. For example, accident sequence $S_2C-\delta$, which dominates PWR release Category 3 (Appendix V, Table V-16), is a small break accident with loss of containment sprays resulting in containment failure by overpressure. According to Appendix VIII of WASH-1400 (Figure 6), the containment failure is calculated to occur in about 200 minutes, while Table VI-2 shows a release time of 2 hours (120 minutes). It is not clear why these values differ.

For the PWR Category 1 release, Table VI-2 of WASH-1400 indicates identical times (1.5 hours) for "time of release" and "warning time for evacuation." Page 32 of Appendix VI indicates (Table VI-7) that a constant time lag (TL) "...associated with interpretation of data and issuance of warning to evacuate" was used for all accidents. It would thus appear that the warning time for evacuation would always have to be 1/2 hour less than the time of release.

The definitions of TL, t_j on page 31 are not clear (eg, what is the difference between "...awareness of impending core melt..." and "...interpretation of data...?") and their relationship to the headings of Table VI-2 is not apparent.

4. Tritium Release Considerations (BWR & PWR)

This section presents an assessment of the radiological hazards associated with the release of tritium during a power reactor LOCA in which core meltdown and containment failure are assumed. The assessment is based on a very conservative, scoping analysis. The risks associated with tritium release during a reactor accident were not considered as part of the Reactor Safety Study (WASH-1400). It is concluded that the radiological hazards associated with the tritium release are negligible compared with the hazards presented by other fission products released during the accident.

Tritium Inventory During Reactor Operation - Tritium exists in the fuel of a power reactor, being produced as a fission product at the rate of one tritium atom for every 1 to 2×10^4 fissions(42). Tritium also exists in the primary system coolant, being generated from neutron reactions with chemical constituents in the coolant (boron, lithium, deuterium). A small amount of tritium produced in the fuel can leak through the fuel pin cladding into the coolant. The amount of tritium expected to exist in the fuel and in the coolant will be treated separately.

(1) Tritium accumulation in the fuel - A recent calculation(43) indicates that the equilibrium amount of tritium existing in a 3558 MWt PWR core is expected to be 11,530 Ci. The amount present in a BWR core should be comparable.

(2) Tritium accumulation in the reactor coolant - The equilibrium amount of tritium in the reactor coolant for a 3300 MWt PWR has been calculated(43) to be 5200 Ci at the end of plant life (40 years), conservatively assuming no letdown and leakage of the primary system coolant. A second calculation(45), done for a 2200 MWt PWR of different design, yields an equilibrium value of about 1000 Ci at a low 6.3 ml/sec (0.1 gpm) primary coolant removal rate. A conservative production rate for a BWR(44) is

0.53 μCi/sec, which would result in an end of plant life inventory, assuming no losses, of 6685 Ci.

For the purposes of this study, the largest amount calculated, 6685 Ci, will be used as the source term.

Biological Effects of Tritium Release During a LOCA - In order to conservatively assess the biological effects of tritium release during a LOCA, the following assumptions are made:

(1) All tritium in the primary system coolant is released to the containment during blowdown.

(2) All tritium in the fuel is released during the core meltdown.

(3) The containment fails immediately after core meltdown, and the entire tritium inventory in the containment (representing 100 percent of coolant inventory plus 100 percent of fuel inventory) is released.

(4) The release occurs at ground level.

(5) The release is assumed to be continuous during the assumed 0.5 hour release time. The 0.5 hour release time corresponds to the shortest release time calculated from the Reactor Safety Study(46).

Under these assumptions, the inhalation dose for a human receptor is(47):

$$D = \frac{2\ AEB\bar{C}\ T}{\lambda e\ M} \qquad (1)$$

where D is infinite (lifetime) dose in rems

A is dose conversion constant

E is average decay energy in MeV/dis

B is breathing rate of the receptor

$\overline{C}$ is the average concentration of tritium in ground level air

T is exposure duration in days

λe is tritium elimination rate for receptor

M is mass of body water

Since tritium is a beta emitter, the whole body dose from the radioactive cloud can be ignored. Also, consistent with assumptions used in WASH-1400 for other isotopes, the dose from ingestion of significantly contaminated vegetation, milk, meat or water is ignored since these sources can be detected and isolated from human consumption. In addition, for the LOCA considered, evacuation would be effected before any significant ingestion of the contaminated foodstuffs could occur.

The numerical values for the terms in equation (1) are as follows:

$A = 5.1\text{x}10^{-5}$ rem - g - dis/pCi-MeV-day

$E = 6.3\text{x}10^{-3}$ MeV/dis

$B = 20\text{m}^3$/day (adult)

5m^3/day (infant)

$\lambda e = 0.069$/day (adult)

0.22/day (infant)

$M = 43{,}000$ gm (adult)

6,100 gm (infant)

Using the above values, equation (1) becomes:

$$D = 4.19\text{x}10^{-9}\ \overline{C}T \text{ - for adult and,} \qquad (2)$$

$$D = 2.39\text{x}10^{-9}\ \overline{C}T \text{ - for infant.} \qquad (3)$$

In the above equation, $\overline{C}$, the ground level concentration, is evaluated from the following equations:

$$\overline{C} = \overline{K}\ Q \qquad (4)$$

where

$\overline{K}$ is the weighted mean dispersion constant (sec/m^3),

and

Q is the release rate in pCi/sec,

$\overline{K}$ has been evaluated$^{(48)}$, assuming average weather conditions, to be:

$\overline{K} = 2\text{x}10^{-6}$ sec/m^3 at 1 Km from the source, and
$\overline{K} = 1\text{x}10^{-7}$ sec/m^3 at 5 Km from the source.

The value of Q for this problem is:

$$Q = \frac{\text{fuel inventory + coolant inventory}}{\text{release time}}$$

or, from the previous discussion,

$$Q = \frac{(11{,}530 + 6685)\text{x}10^{12}}{0.5(3600)} = 1.0\text{x}10^{13}\ \frac{\text{p Ci}}{\text{sec.}}$$

Thus,

$$\overline{C} = (1.0\text{x}10^{13})(2\text{x}10^{-6}) = 2\text{x}10^{7}\ \text{pCi/m}^3 \qquad (5)$$

at a distance of 1 Km.

Substituting the value of $\overline{C}$ from equation (5) into equations (2) and (3) yields:

$$D = 8.4\text{x}10^{-2}\ \text{T rem (adult)} \qquad (6)$$

and

$$D = 4.8\text{x}10^{-2}\ \text{T rem (infant)} \qquad (7)$$

Assuming an exposure time of one day, the doses become

D = 84 mrem (adult)

and

$$D = 48 \text{ mrem (infant)}.$$

Assuming the worst weather conditions (Class F), the ground level concentration can be(48) a factor of seven higher, giving dose rates of

$$D = 588 \text{ mrem (adult)}$$

and

$$D = 336 \text{ mrem (infant)}.$$

It has been observed that tritiated water vapor can be absorbed through the skin and produce an internal dose effect up to as large as that obtained from inhalation(59,60). Assuming, conservatively, that such an effect occurs, the doses would be doubled, and would become

$$D = 1076 \text{ mrem (adult)}$$

and

$$D = 672 \text{ mrem (infant)}.$$

Conclusions - The doses at 1 Km during Pasquill Class F weather due to 100 percent tritium release from the reactor coolant and fuel during a LOCA are insignificant (1076 mrem - adult and 672 mrem - infant) when compared with doses from other fission products at the same location. It requires, according to WASH-1400(46), 3,000 to 5,000 rem to the lung to produce an acute fatality, or an adult dose $2.7\text{x}10^3$ to $4.5\text{x}10^3$ times larger than those calculated due to the tritium release. The incidence of lung cancer fatalities due to radiation is 1-2 deaths per year per million man-rem(46). Thus, for the adult doses computed, an exposed population at 1 Km of about 850,000 adults would be required before any lung cancer deaths would be expected from the tritium released during the assumed accident. Similar comparisons could be made for the whole body dose or other organ doses from tritium.

VI. GENERAL OBSERVATIONS

While the main thrust of this report is to present the detailed analysis of specific areas in WASH-1400 in an attempt to determine their applicability, it is considered at least as important to discuss and explore some of the more fundamental, far-reaching implications and limitations of the document. Current indications are that WASH-1400 could become the single most important document forming the basis for decisions in this country regarding the safety of nuclear power for generating electrical energy. It is essential, therefore, that the report present the safety risks in a comprehensive and unambiguous manner, using the best available scientific methods and technology. It is in this spirit that the following observations are presented.

A. COMPARATIVE RISK CURVE

The net result of the WASH-1400 analysis is depicted in Figure 29. This curve appears in the summary report of WASH-1400 (Figure 1, page 3), and in the main document (Figure 6.1, page 189). It shows the frequency of accidents versus the number of fatalities for several man-caused events including accidents in 100 nuclear power plants. This curve has been widely published to show that the risks to the general population from nuclear power plant accidents is much less than from any other man-made cause examined. In view of the significance of this curve, and its widespread use to convey the implication to the public that reactors are safe, it is imperative that the comparison which it depicts is valid. In reviewing the basis for the curve, several factors were found which tend to undermine the credibility of the comparisons. (These factors are in addition to the more detailed technical problems discussed in Section V.) The factors are as follows:

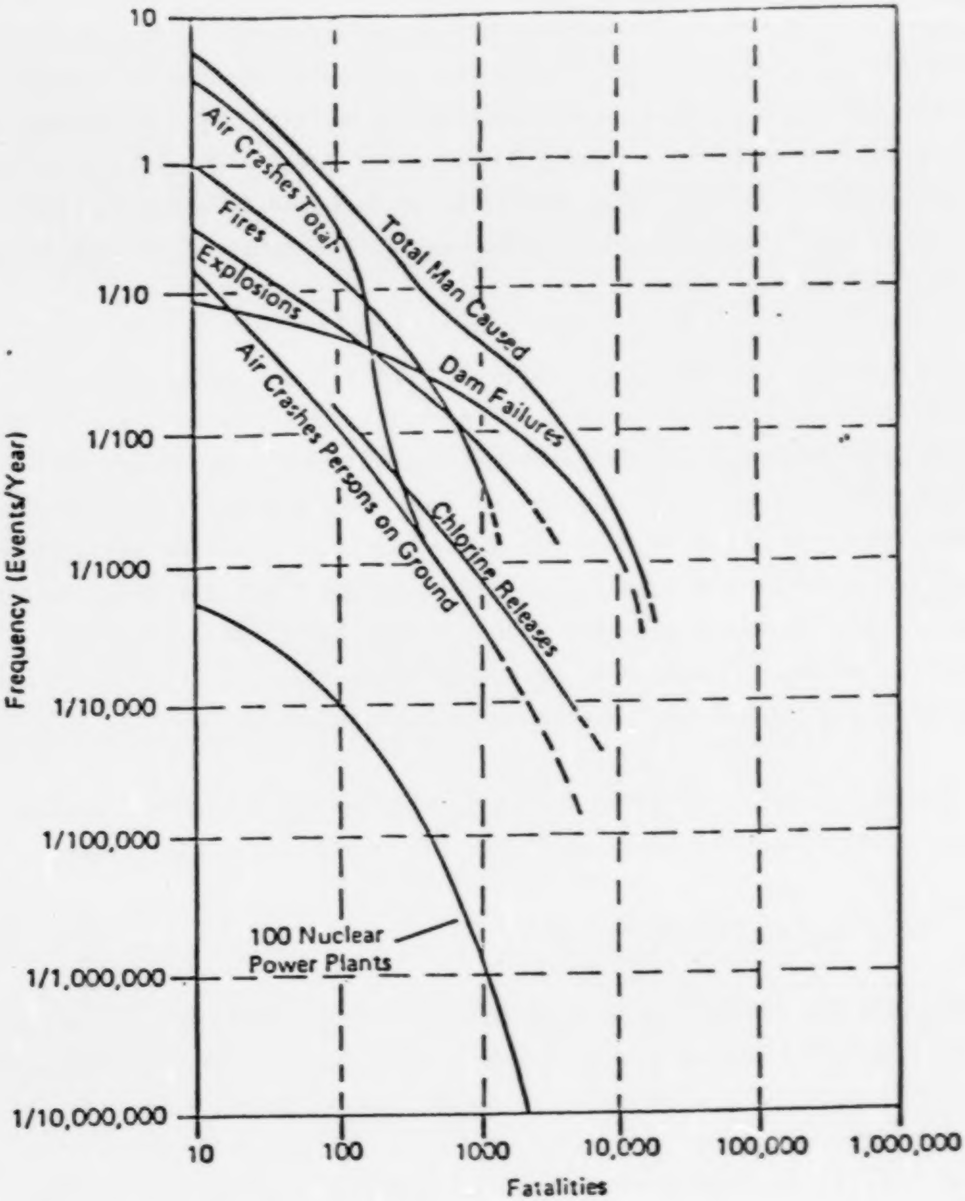


Figure 29 - Comparative Risk Curve from WASH-1400

1. Calculated versus Actual Risks

Figure 29 from WASH-1400 compares nuclear power plant risks calculated in WASH-1400 with actual risk data from the other man-caused events. An important question is, obviously, how closely do the calculated risks compare with actual risks from nuclear power plants? The curve is misleading, in that risks appear to be compared on a common basis, when in fact actual risks from other causes are compared to calculated risks from nuclear power.

An interesting feature of the curve is that all man-caused risks, except nuclear, fall in an envelope of width less than a factor of 100 (frequency value for a given number of fatalities) even though the risks come from extremely diverse sources. The nuclear risks, on the other hand, fall over two decades below the lower limit of the envelope. This large difference in risks is obviously not a valid reason, in itself, for disputing the WASH-1400 results, but it does raise the question of whether the curve is a valid comparison of risks or illustrates how uncertain nuclear risk calculations are.

At the very least, the curve should clearly indicate that the nuclear power plant risks are calculated risks as opposed to actual risks.

2. Calculational Uncertainties

Throughout WASH-1400, frequent reference is made to the uncertainties in the various calculations, and numerical bounds are stressed on most results. However, in Figure 1 of the WASH-1400 summary report (reproduced as Figure 29) no such uncertainty bounds are displayed in the curve, nor is any hint given in the accompanying text as to the confidence limits of the results shown. Uncertainties in calculations, when estimates can be made, should always be clearly indicated in the results. Figure 6.1 on pages 189 and 227 of the main document of WASH-1400 is essentially the same as Figure 29 and, here again, no uncertainties in the calculations are indicated. Pursuing further, the source of Figure 29, Figure 5.3 (main document, page 153 of WASH-1400) appears as the precursor to the risk comparison curve. This curve is reproduced as Figure 30. Here, in fine print at the bottom, the uncertainties are quoted. Applying these uncertainties to Figure 30 and translating them to Figure 29 yields Figure 31. As shown, the uncertainty is rather large, and is biased in the direction of increased risk. Both of these factors tend to amplify the need for clearly showing the uncertainties in this important curve. Displaying this uncertainty also gives a strong indication that the comparisons are made on a different basis; ie, the nuclear risks are calculated and the others are from actual data (see preceding comment).

3. Acute versus Total Fatalities

Figure 29, extracted from WASH-1400, shows fatalities versus frequency of events for 100 nuclear power plants. The implication given is that these fatalities are the total deaths expected as a result of the nuclear accident. However, these fatalities from 100 nuclear power plants are acute fatalities derived directly from Figure VI-5 from Appendix VI, page 72 of WASH-1400. (This curve is the same as that included in the main document, page 153 of WASH-1400.) This curve is reproduced as Figure 30. (Figure 29 is obtained by multiplying the ordinate of Figure 30 by 100 to account for 100 nuclear plants.) Figure 30 is clearly labelled "Acute Fatalities." For some reason, the word "acute" was unfortunately dropped from the abscissa in Figure 29. "Acute" is used to describe deaths which occur immediately after, and as a result of, the accident. One unique characteristic of nuclear power plant accident calculations is the large number of calculated latent deaths (occurring over a 20-year period following the accident) caused by the release of radioactivity. Essentially all of these latent deaths are the result of cancer induced by exposure to radioactivity from the nuclear plant accident. According to WASH-1400, a value of 100 latent deaths per 10^6 man-rem was used to compute the

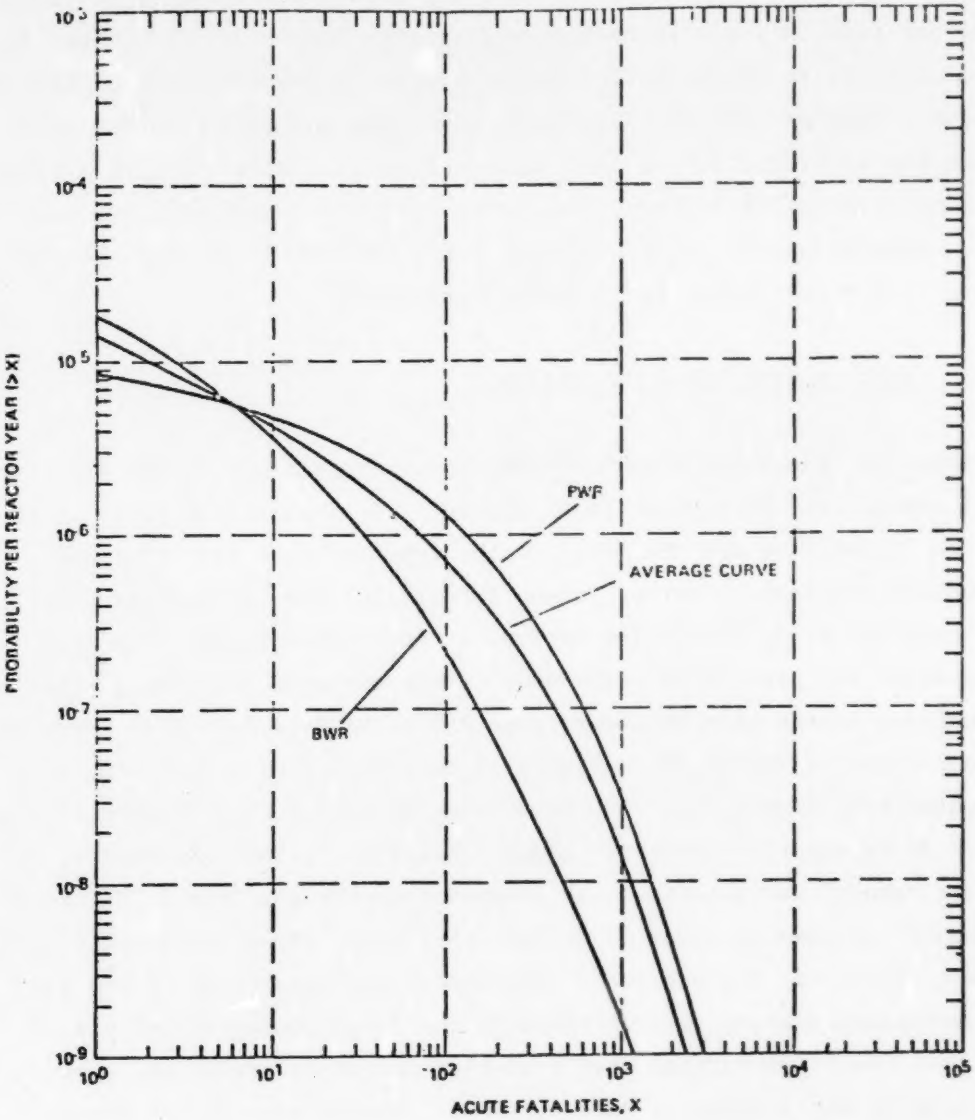


Figure 30 - Acute Fatality Curve from WASH-1400

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* Uncertainties can be accomodated by allowing variations in the average curve of factors of 1/3 and 6 in probability and 1/3 and 3 in consequence

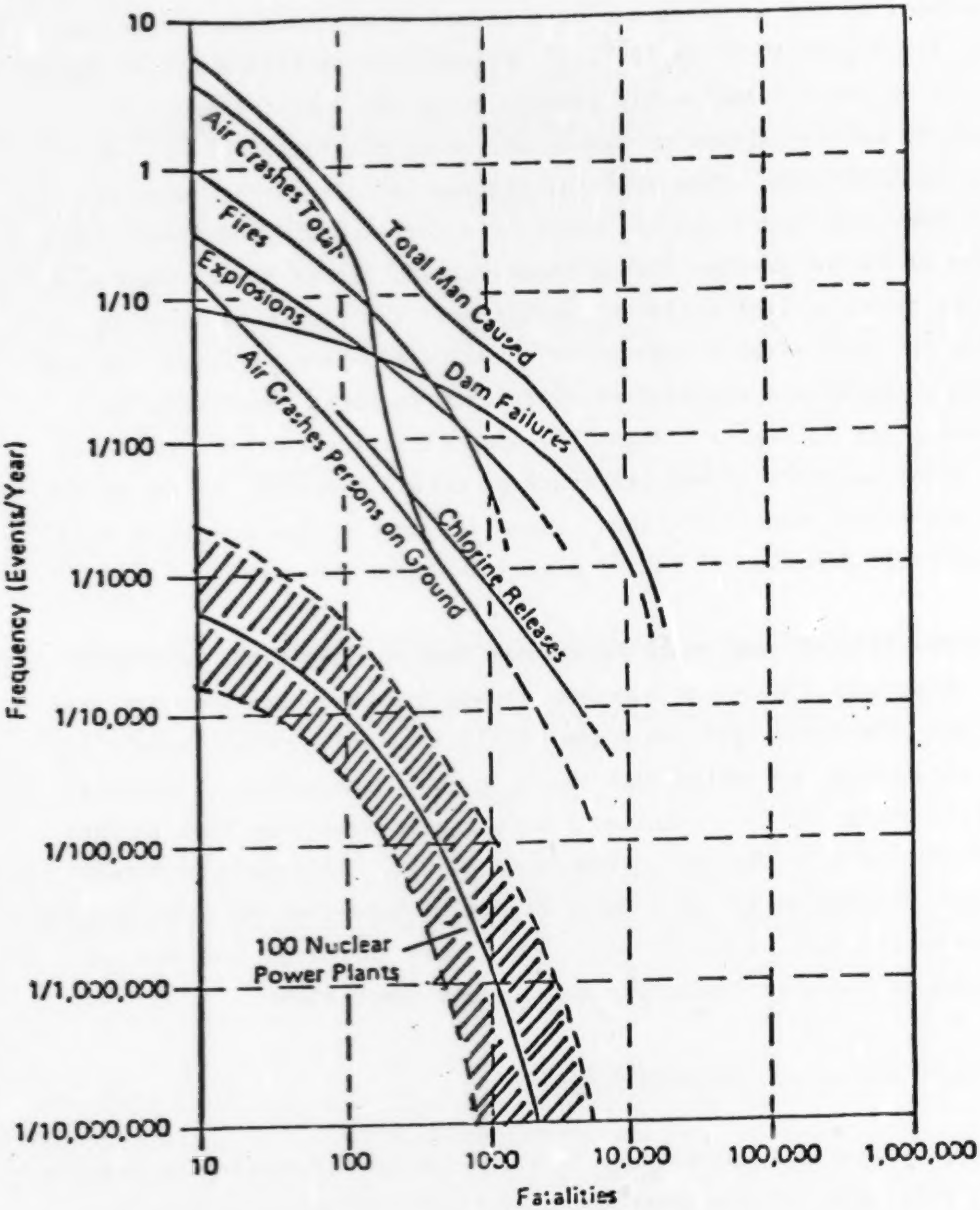

Figure 31 - Comparative Risk Curve
with Uncertainties Shown
as Quoted on Pg. 153 of
Main Document (WASH-1400)

latent deaths from the nuclear accidents (Appendix VI, page 34). To determine the total number of latent deaths, Appendix VI (page 73) states, "The number of latent deaths due to cancer over a 20-year period subsequent to the accident can be obtained by multiplying the bottom scale (of Figure VI-7) by 10^{-4}..." Figure VI-7 is reproduced as Figure 32. To obtain a relationship between acute and latent fatalities, Figure 32 was translated to Figure 30, using the factor of 10^{-4} as given in WASH-1400. This produces Figures 33. As can be seen, for every number of acute deaths, there is a corresponding, and much larger, number of latent deaths. Using these curves, Figure 31 was redrawn to include total nuclear accident fatalities. The result is shown in Figure 34, with error bounds given in WASH-1400 also included. As can be seen, there is a significant difference in total fatalities for nuclear plant accidents. WASH-1400 does not state whether fatalities from other man-made causes are acute or total. However, it should not make any significant difference, since for most of the accidents used, a very small fraction of latent deaths would be expected.

The comparison of only acute fatalities from nuclear plant accidents with fatalities from other man-made causes is considered somewhat misleading. The total risk, in terms of all fatalities, would appear to be a significant parameter with which to judge the safety of nuclear power, although it is recognized that the perceived risk from latent deaths may be less than that from acute deaths. It should be noted that the discussion in the Summary Report of WASH-1400 does not include latent deaths, but uses only acute fatalities (although they are not labelled as such) in discussing the nuclear plant risks.

4. Extrapolation of Man-Caused Risks

The comparative risk curves used in WASH-1400 to illustrate the results of the risk calculations (reproduced as Figure 29) shows several man-caused risks. The basis for these plots is derived from the analysis contained in Chapter 6 of the main WASH-1400 document. The solid

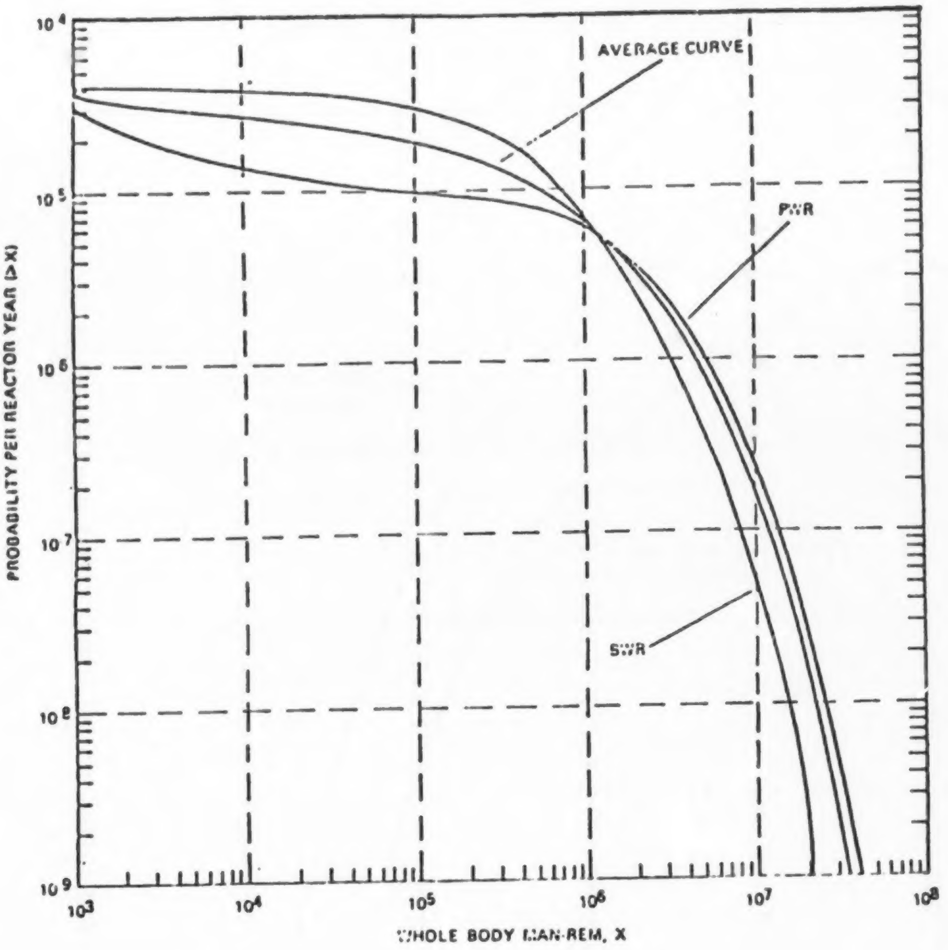

Figure 32 - Whole Body Man-Rem Curve from WASH-1400

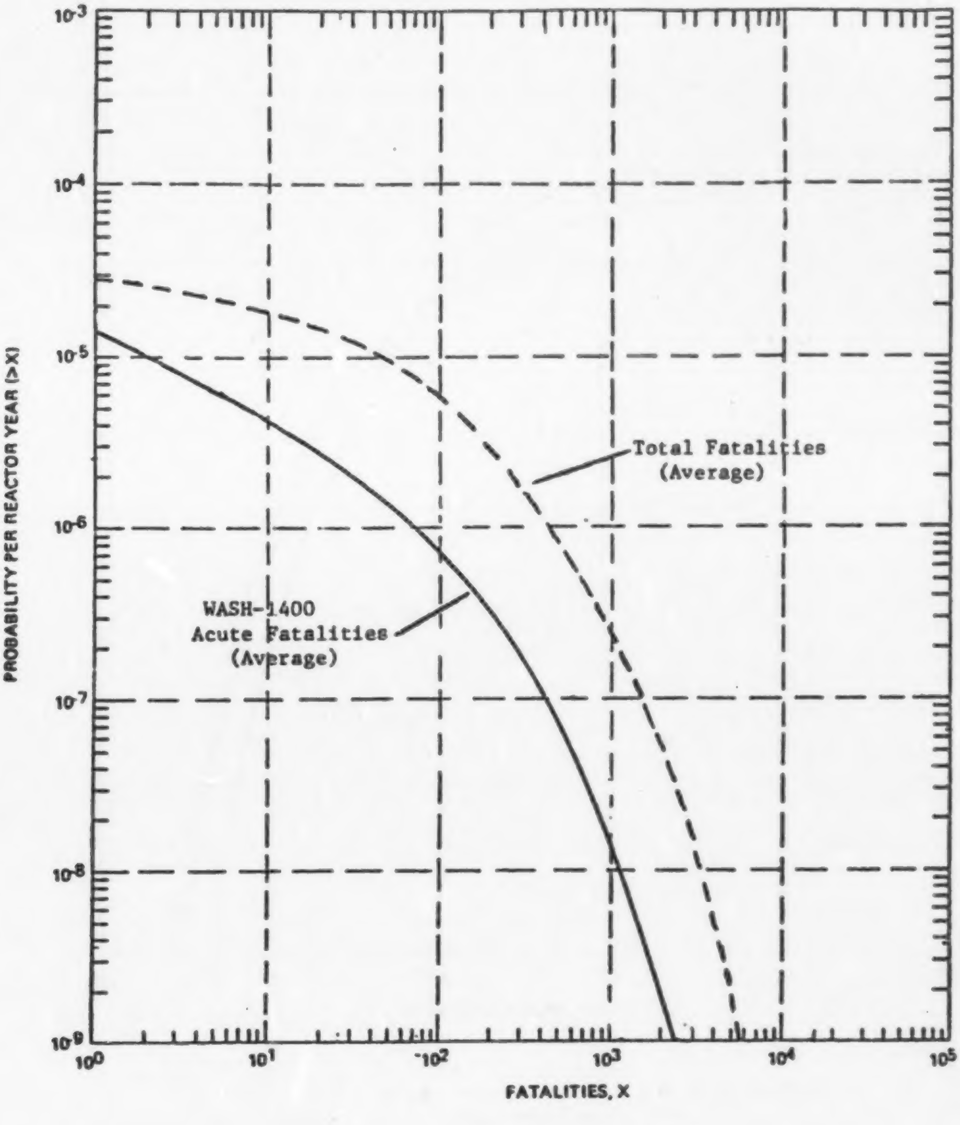

Figure 33 - Acute vs. Total Fatalities

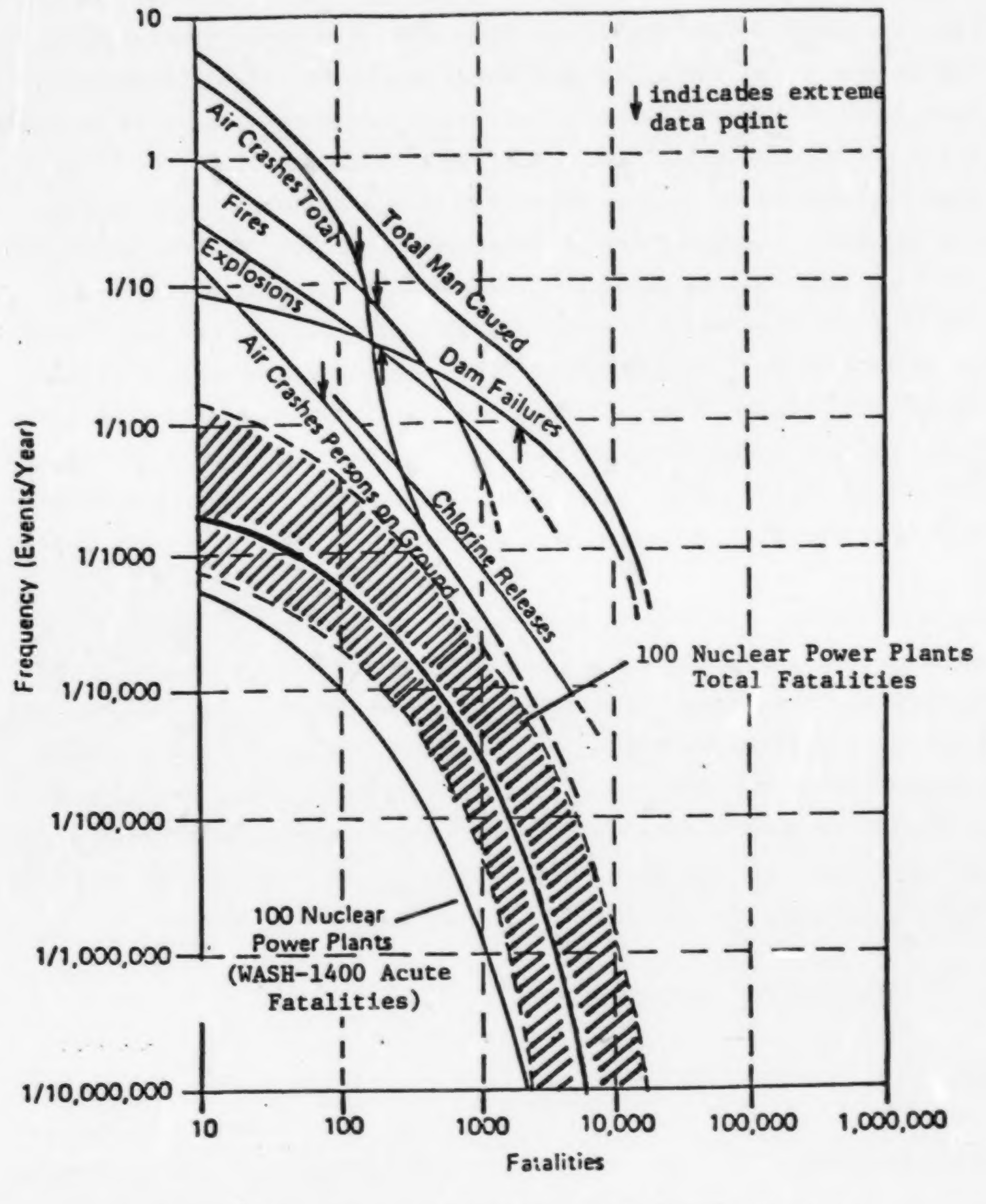

Figure 34 - Total Fatalities, with Uncertainties, Using Figure VI-7 of Appendix VI, WASH-1400

curves extend out to some fatality value, then become dashed before terminating. It is not clear what the dashed portions of the curves imply. The usual implication is that the dashed portion indicates an extrapolation beyond the point where the data ends. However, in the case of the WASH-1400 curve, the data ends at a value considerably before the end of the solid portion of the curve. The extrapolated portion of the curves is based on various arguments presented in Chapter 6 which necessarily have a different basis for each type of risk. Some extrapolations appear valid, others are somewhat questionable. For example, the discussion of fatalities from dam failures given in the WASH-1400 main document concludes with a calculated frequency (10^{-3}) for a dam failure causing 10,000 fatalities. This calculated value is said to "...agree quite well with the extrapolation of the data," and a curve (Figure 6.10, page 216) is shown to illustrate the point. The calculated point does not, in fact, agree well at all with the extrapolation of the data. The data argue for an extrapolation with the opposite inflection than was used.

Figure 34 shows the extreme data points for each curve, indicated by an arrow. As can be seen, a significant extrapolation was done for each of the man-caused risk curves, particularly for air crashes. (No data was quoted in Chapter 6 for chlorine releases.) Each of these extrapolations has some uncertainty associated with it, and it would be useful to present such uncertainty on the curve. As a minimum, the dashed portion of the curve should start at the extreme data point to clearly indicate the point at which extrapolation commences.

B. NUCLEAR PLANT CHARACTERISTICS

According to WASH-1400, the calculated risks apply to the 100 nuclear power plants expected to be in operation by 1980. Indeed, recent revised estimates(4,5) confirm that 100 power plants are scheduled to be on-line by 1980. However, several characteristics of these plants are different than those assumed by WASH-1400. These differences are:

1. Distribution of Plant Type

According to recent(4,5) estimates, by 1980 there will be 33 BWRs and 67 PWRs operating. Thus, the total risks from all plants must be weighted two-to-one towards PWRs. WASH-1400 uses an unweighted average to determine the risks from these plants in arriving at the final risk assessments. This is done for acute fatalities, acute illnesses, whole-body man-rem, property damage, etc (Figures VI-5 through VI-9 of Appendix VI). To determine the difference between such unweighted averaging and the actual case accounting for the preponderance of PWRs, the acute fatalities, as presented in Appendix VI (Figure VI-5, page 72) and reproduced here as Figure 30, were adjusted. Figure 35 presents the results of adjusting the average risk to account for the projected existence of twice as many PWRs. Since the risks are generally about the same for the two plants, not much change is evident in the biased averaging. However, a detailed examination shows some differences which may be significant depending on how the results are used. For example, an accident which has a probability of 10^{-6} would result in an average of 70 acute fatalities according to the WASH-1400 curve, compared to 86 with the revised curve, a 23 percent increase. A somewhat lower fatality number occurs below an accident probability of about $6x10^{-6}$.

Even though, in a gross sense, the change does not appear significant, it is not good scientific practice to compute unweighted averages when it is obvious that weighting is called for.

2. Power Levels

According to Appendix VI (page 68) of WASH-1400, the risks were calculated assuming all 100 plants operate at 3200 MWt. The actual average power level for the 33 BWRs scheduled to be operating in 1980 will be 2400 and for the 67 PWRs, 2650 MWt(5). Since accident consequences are

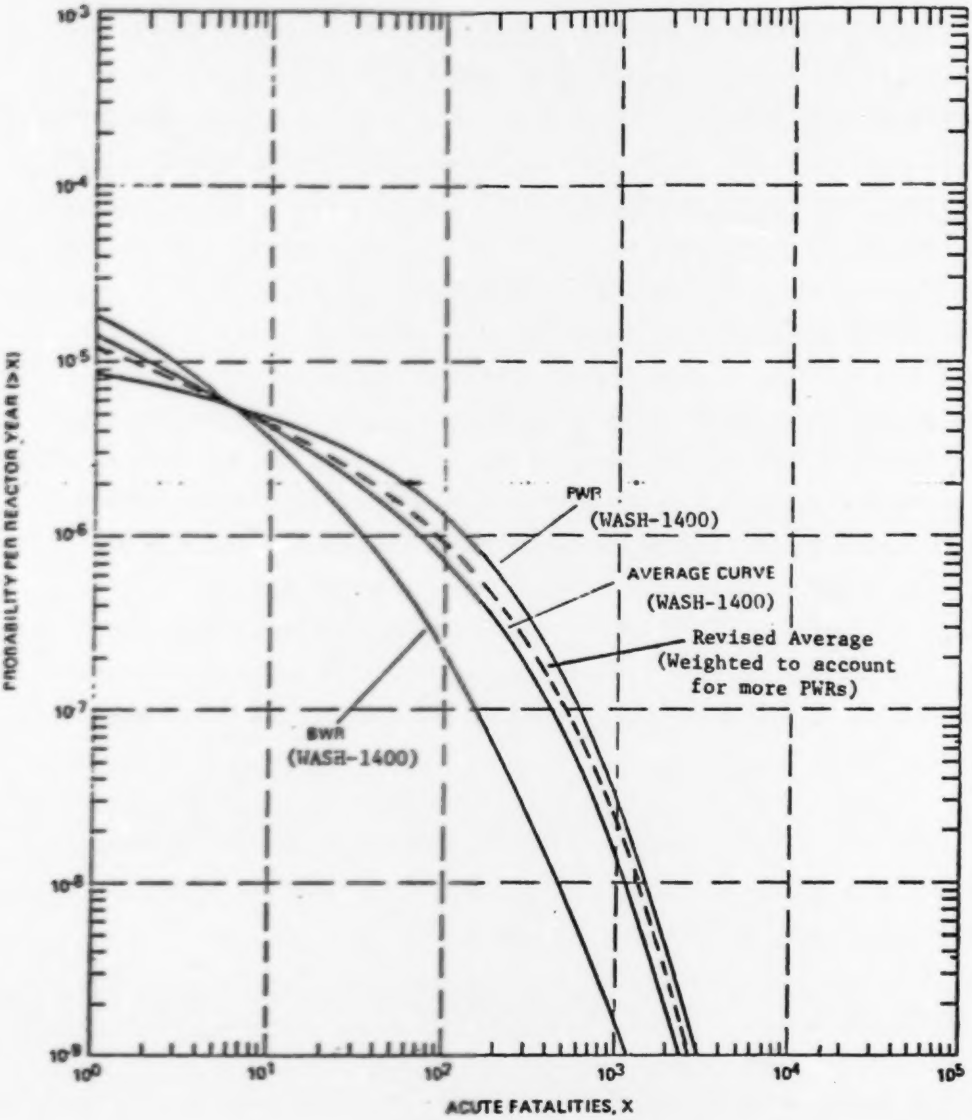

Figure 35 - Acute Fatalities Showing the Effect of More PWRs than BWRs

directly related to the amount and type of radioactivity released, which is, in turn, a linear function of power level, a reduction in assumed power level should have a proportional reduction in accident consequences. Figure 36 shows the reduction in acute fatalities from the WASH-1400 numbers when the computed reductions are accounted for (the results are also weighted to account for the more numerous PWRs as discussed in "A" above), conservatively assuming, as was done in WASH-1400, that all of the plants run continuously at full power. The difference does not appear very significant, although the average acute fatalities are reduced from 400 to 300 for an accident whose probability is 10^{-7}. Also, the average maximum fatalities are reduced from 2300 to about 1700, a 25 percent reduction. Again, these seemingly insignificant differences can, under certain circumstances, become important. It should be noted that the most severe accident consequences calculated won't change since there will be 3200 MWt reactors operating in 1980.

3. PWR Design Variations

According to Reference 4, the PWRs expected to be on-line by 1980 are distributed, by vendor (and by number of primary loops in the case of Westinghouse designs), according to Table 19.

The WASH-1400 PWR risks are computed based only on the Surry reactor which is a Westinghouse 3-loop design(3), representative, according to Table 19, of 21 percent of all PWRs scheduled to be operating by 1980. Risks from other PWR types are not computed in WASH-1400, and an assessment of the effect of considering the more numerous Westinghouse 4-loop reactors (represented by the Trojan reactor(2)) is analyzed elsewhere in this report. Not considered, either herein or in WASH-1400, are the effects on PWR risks represented by the Combustion Engineering and Babcock & Wilcox designs (representing a total of 34 percent of the plants). These designs are different from the Westinghouse plants (see, for example, References 49 and 50), and it cannot be assumed,

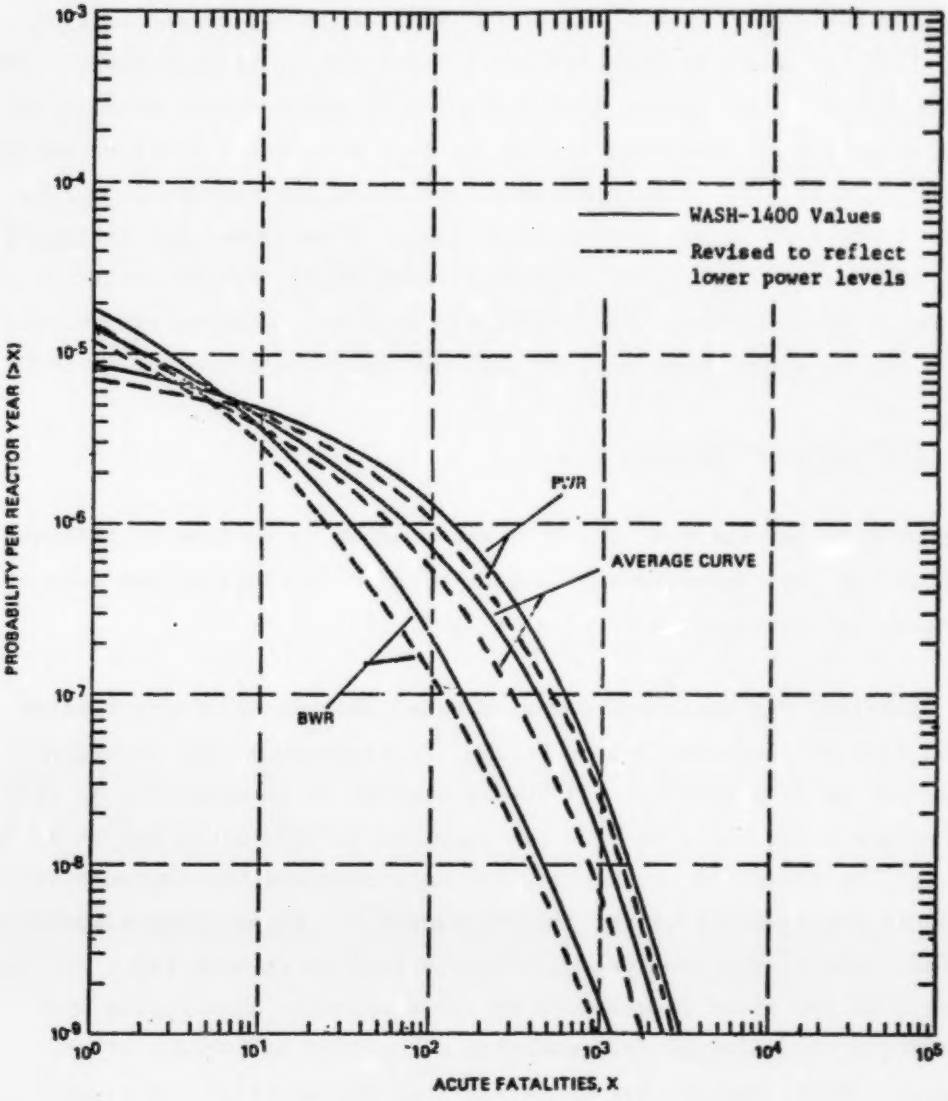

Figure 36 - Effect of Reducing Average Power of BWRs from 3200 MWt to 2400 MWt and PWRs from 3200 MWt to 2650 MWt

Table 19 - DISTRIBUTION OF PWRs EXPECTED TO BE OPERATING BY 1980

Reactor Vendor	Type	Number	% of Total
Westinghouse	2-loop	8	12
Westinghouse	3-loop	14	21
Westinghouse	4-loop	22	33
Combustion Engineering	*	10	15
Babcock & Wilcox	*	13	19

* Only one basic type is marketed by Combustion Engineering and Babcock & Wilcox.

a priori, that the Surry risk analyses can be applied across the board to these plants as was done in WASH-1400.

C. REALISTIC VERSUS CONSERVATIVE ASSUMPTIONS

Page 15 of the main report in WASH-1400 contains a list of "specific objectives" which were added to the original charter. The second of these states that the Reactor Safety Study will "perform a more realistic assessment (of nuclear power plant risks) as opposed to the 'conservatively oriented' safety approach taken in the licensing process for nuclear power plants." While this is a laudable approach, it does not seem to have been consistently followed. In many cases, the report contains assumptions which are described as conservative. In most instances, the effect of the assumption is negligible, and the authors appear to have fallen into the tempting position of making conservative assumptions, unlikely to be challenged, in lieu of more extensive analysis when the assumption does not have a deleterious effect on the result. The pitfall in this approach is that special applications of the results or alterations of the analysis due to new data or techniques can cause insignificant conservative assumptions to become significant conservative assumptions.

There are many areas where, due to inadequate analytical techniques and/or lack of data, conservative assumptions may be justified, and WASH-1400 did appropriately use conservative assumptions in many cases. In at least one case, however, WASH-1400 makes an acknowledged conservative assumption which, based on analyses conducted in Section V, Item 1 of this report, could be significant. In this case, WASH-1400 made the "extremely conservative" (Appendix II, Vol. 3, page 94) assumption that any three adjacent BWR control rod insertion failures during scram would not render the core subcritical. It seems likely that, in this case, information is available from which to make a realistic assumption. As discussed in Section V, Item 1, this can become a very crucial assumption, significantly influencing the ultimate BWR public risks.

Realistic assumptions, with well justified error bounds, should be used whenever possible in an important scientific work such as WASH-1400.

D. COMPARISON OF RISKS BETWEEN NUCLEAR AND OTHER MEANS OF ELECTRICAL POWER GENERATION

The WASH-1400 comparisons of risks between nuclear power generation and risks from other selected man-caused sources is one important aspect in judging the acceptability of nuclear power. However, there are other considerations that must be evaluated in arriving at a final judgment, one of the more significant of which is the comparison of risks from other means of generating electrical power, an issue not considered in WASH-1400. Such a comparison would allow a determination of which means of generating electrical power imposes the least additional risk on the general population, an issue quite different from the comparison between nuclear power risks and other man-caused risks.

In this regard, Chapter 13 of Reference 57 assesses some of the risks associated with electrical power generation from coal-fired plants.

It should also be noted that the risks calculated in WASH-1400 for 100 nuclear power plants include only risks from reactor accidents in the power plants. Risks from uranium mining, fuel reprocessing, spent fuel transportation, sabotage, Pu diversion, radioactive waste transportation and storage, etc, are not included. WASH-1400 does point out initially that the report covers only risks from reactor accidents. However, the risk curves are merely entitled "100 Nuclear Power Plants." Some readers may erroneously conclude that the curves represent the total risks associated with the operation of 100 nuclear power plants. The risk associated with all fuel and waste handling activities required to support the operation of these plants is not included.

E. GENERAL INCONSISTENCIES

WASH-1400 contains many inconsistencies including variations in approach and different levels of depth in detail considered. Many of these are mentioned in Section V. Although these inconsistencies do not necessarily invalidate the results, they do tend to undermine the confidence the reader obtains in the credibility of the results. These inconsistencies should be minimized.

F. CONCLUSIONS

WASH-1400 represents the most significant effort ever attempted at quantifying the public risks associated with nuclear power. Although modest attempts have been made(51) similar to the WASH-1400 technique of quantifying reactor risks, the depth and breadth of the Study is unprecedented. Although not a new idea (such a study was suggested by Pugh(52) in 1969), the Study contains many innovations that substantially improve the perspective of power reactor safety. The so-called "maximum credible accident" approach used for many years by the AEC has clearly been shown to be an oversimplified method for evaluating nuclear safety under current conditions (as also pointed out by Pugh(52)). Weak spots in both accident analysis and safety system design have been identified by the Study, and this in itself is a major accomplishment. Despite some technical problems, as discussed in this report, the Study still represents the most profound and far-reaching assessment ever attempted in an effort to describe power reactor risks.

The Study cannot remain static, however. Continuous, diligent effort must be applied to keep the study up to date by timely incorporation of reactor design changes, new designs, changing failure rates, etc. Only in this manner can undesirable trends be perceived and corrected. The Study must also be expanded to incorporate different PWR designs as well as different power reactor concepts. Extending the Study to apply beyond 1980 should be started. Only by continuous attention to updating the document can a current assessment of power reactor risks be maintained.

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VIII. GLOSSARY

AEC	Atomic Energy Commission
BWR	Boiling Water Reactor
PWR	Pressurized Water Reactor
RPS	Reactor Protection System
LPIS	Liquid Poison Injection System, Low Pressure Injection System
GE	General Electric Company
LOCA	Loss of Coolant Accident
LPCI	Low Pressure Coolant Injection
RHR	Residual Heat Removal
ESF	Engineered Safety Feature
HPIS	High Pressure Injection System
SIS	Safety Injection System
RWST	Refueling Water Storage Tank
LPRS	Low Pressure Recirculation System
RSS	Reactor Safety Study (WASH-1400)
SAR	Safety Analysis Report
EPS	Electric Power System

APPENDIX A

A. INTRODUCTION

This Appendix describes Subtask I of the WASH-1400 review, which consisted of selecting items which were to be subsequently given an in-depth review. The criteria for selection of the items is presented, as well as the specific reason that each selection was made. The items are divided into two categories. The first category consists of failure mode paths and the second category includes critical consequence areas. The items selected, as described in this Appendix, were subsequently modified as the in-depth review proceeded. The modifications, and the final items selected, are described in Section III (Introduction) of this report.

B. DEFINITION OF FAILURE MODE PATHS AND CRITICAL CONSEQUENCE AREAS

For the purposes of this Study, failure mode paths are defined as accident sequences involving only the primary system and directly associated subsystems which are calculated to culminate in core meltdown. Critical consequence areas are those areas directly associated with the time sequence and magnitude of radioactive release from the containment building following or during a failure mode path event. Thus, failure mode paths are paths or sequences, following an accident initiating mechanism, in which the thermal-hydraulic response of the reactor core is of primary concern. Critical consequence areas, on the other hand, refer to the disposition of fission products released from the primary system during the failure mode path event.

Failure mode paths are characterized in WASH-1400 by event trees which culminate in core melt. The fault trees used to compute the probability of each failure in the event tree are also considered part of the failure mode path. The response of the containment and the attendant disposition of radioactive material is considered separately in WASH-1400. Thus, the definitions used here are consistent with the general format of WASH-1400.

Failure mode paths which were considered for selection can be categorized as one of the following:

(1) An accident sequence (path) which was overlooked and thus not considered by the Study. This sequence could include an overlooked initiating mechanism or an overlooked failure mode path following a given initiating mechanism.

(2) An accident sequence incorrectly analyzed in the Study. This could include faulty event tree logic, or incorrect application or manipulation of failure probabilities.

(3) A protection system malfunction overlooked. This includes an omission in the fault tree analysis of protection system functions. Overlooked common mode failures are also included.

(4) An incorrectly analyzed protection system function. This includes incorrect logic on protection system fault trees or incorrect application or manipulation of component failure probabilities.

Similarly, critical consequence areas can be categorized as:

(1) Parameters affecting containment leakage or failure sequences which were overlooked in the Study,

(2) Parameters in the containment leakage or failure sequence which were incorrectly analyzed,

(3) Containment protection system failure sequences which were overlooked, and

(4) Containment protection system fault trees which were incorrectly analyzed.

C. CRITERIA FOR SELECTION OF FAILURE MODE PATHS

The following criteria were used in selecting the failure mode paths:

(1) The path must appear to have a significant impact on the results of the Study. This applied both to overlooked paths and to apparently incorrectly analyzed paths.

(2) If a path from the Study is selected, it must appear to be incorrect on the basis of either some apparent problem in the internal logic or analysis of the path or on a judgment of the validity of the result.

D. FAILURE MODE PATHS SELECTED

The following failure mode paths were selected for an in-depth review. A discussion of the reasons for the selection is included based on the Selection Criteria presented in Section III.

1. BWR Reactor Protection System (refer to Appendix II, Vol. 3, Section 6.2 of WASH-1400)

This system constitutes a protection system designed to mitigate numerous accident sequences. As a failure mode path, it falls in Category 4 of Section II. The selection of this system is based on the following factors:

(a) The computed failure probability of the system, as presented in the Study, appears to be unreasonably low.

(b) The system is required for a wide range of accidents. In particular, it is required during high probability transient events and its failure could lead to severe consequences.

(c) Credit is taken during the more likely transients for actuation and operation of the standby liquid control system to deliver sodium pentaborate solution to the reactor coolant system and thus affect reactor shutdown. The ability of this system to be effective in time to arrest many transients is questionable, and its use in this regard appears to be in conflict with an AEC-Regulatory assessment(1).

2. BWR Anticipated Transient #1

This transient accident, as well as the next two (see below) will be selected later in the effort based on the following:

(a) The transient selected will be one which is both likely and imposes the most stringent requirements on the reactor protective system. This latter factor cannot be assessed until the protective system is examined in detail as proposed under item 1 above.

(b) The transient selected will be one which was not considered by the Study and which meets the requirements of (a) above.

3. BWR Anticipated Transient #2

See 2 above.

4. BWR Anticipated Transient #3

See 2 above.

5. PWR - Electric Power System Availability

The availability of this system during all major accidents is critical to the mitigation of the accident consequences. The failure probability computed in the Study appears to be low for such a complex system. Thus, the PWR electric power system has been selected as a Subtask II review.

6. PWR - High Pressure Injection System (HPIS)

This system was selected since it must operate during the relatively high probability small break accident condition. There is no backup system to the PWR-HPIS (as there is in a BWR); thus, its failure strongly influences the overall consequence assessment. Some aspects of the system, such as the apparent necessity for operator action to actuate the pump lubrication system may not have been accounted for.

7. PWR - Small Break LOCA #1

This accident has been selected because of the following considerations:

(a) The accident has a relatively high probability,

(b) It appears that the Study may have overlooked and/or improperly treated specific small break locations, and

(c) The consequences of a small break LOCA, as indicated by the Study, are a dominating factor in the overall consequences.

The specific Small Break accident, in terms of break size and location, has not been selected. The selection will be based on the following:

(a) The break size and location which impose the most stringent requirements on the HPIS (item 6 above).

(b) The break size and location which results, assuming failure of the HPIS, in the most severe consequences.

8. PWR - Small Break LOCA Transient #2

See item 7 above.

9. PWR - Loss of Power Transient

This accident was selected because it may represent a relatively probable event (see discussion under item 5 above), and because the consequences can be severe. It appears that some events important to the accident consequence, such as discharge of accumulators after core melt-through, may have been omitted.

10. Component Failure Rates

Although not specifically a failure mode path, this activity has been selected since it provides the basis for the system failure rates being reviewed. As presently envisioned, this activity will apply to several of the systems investigated under Subtask II. It will consist of an independent evaluation of failure rates for those components which have been identified as critical to the determination of system failure rates.

11. PWR - Low Pressure Injection System (LPIS)

This system was selected for the following reasons:

(a) The successful operation of the LPIS is required to prevent core meltdown for all LOCAs with a pipe break greater than ∿ 6-inch diameter.

(b) There is no backup system to the LPIS.

(c) The failure probability of the LPIS, as calculated in WASH-1400, appears to be too low.

12. PWR - Low Pressure Recirculation System (LPRS)

This system was selected because:

(a) The successful operation of the LPRS is required to prevent core meltdown for all LOCAs with a pipe break greater than ∿ 6 inches in diameter

(b) There is no backup system to the LPRS.

(c) Successful operation of this system depends on operator action, and it is not clear if this activity has been properly accounted for by the Study.

E. SELECTION CRITERIA FOR CRITICAL CONSEQUENCE AREAS

The following criteria were used in selecting the critical consequence areas:

(1) The area must significantly influence either the magnitude of the radioactive source or the time of release of the source from the containment to the environs, or both.

(2) The area, as indicated by the Study, must either be

   (a) incorrectly quantified,

   (b) improperly applied,

   (c) judged significant but not considered by the Study.

F. CRITICAL CONSEQUENCE AREAS SELECTED

The following critical consequence areas have been selected for further investigation in Subtask III of the contract. A discussion of the reasons for the selection is included, and is based on the selection criteria presented in Section V.

1. Core Parameters Prior to Meltdown Calculation

The core parameters of fuel and clad temperature and water level are critical initial conditions for subsequent calculations of the time to core melt which, in turn, establishes the time of release of radioactivity from the core to the containment. It appears that the assumptions used in establishing these parameters are questionable, and other values for these parameters, which may be more likely and appear to have the potential of causing a significant change in the time to core melt, have not been considered.

2. Core Meltdown Calculations

At the time of fuel melt, large quantities of radioactive material are released from the core and become available for release to the containment. Thus, the time of core melt becomes a significant parameter in determining the radiological consequences of the accident. For this reason, this calculation has been selected for review. In addition, some of the methods utilized in the Study to compute heat transfer from the core appear to be questionable. This task includes an independent calculation of core heatup, utilizing available applicable computer codes.

3. Containment Response-Pressure History (PWR)

The pressure response of the containment following a LOCA blowdown is critical in determining accident consequences. Containment leakage and containment rupture, which are determined by containment pressure response, provide the only mechanisms for airborne release of radioactivity outside the containment. The PWR containment pressure histories appear to be low when compared with similar calculations for other PWR plants. For these reasons, the PWR Containment Response-Pressure History has been selected as a Subtask III critical area.

4. Containment Response-Failure Pressure (PWR)

The pressure at which the containment is assumed to rupture is a critical parameter in the accident consequences analysis since the rupture event causes a puff release of fission products residing in the containment. Since the containment pressure buildup is a time-dependent parameter, the assumed containment failure pressure can have a significant influence on the time of fission product release. In addition, the fission product inventory available for release depends on the residence time of the fission products in the containment vessel, since plateout, washout, and radioactive decay are all time-dependent phenomena.

5. Applicability of Containment Response to Other PWR Systems

This item concerns the applicability of the containment response analysis contained in the Study to more recent PWR designs. Specifically, the results do not consider the PWR ice condenser and fan cooler containment designs. The containment response and failure modes could be significantly different for these designs.

6. Tritium Release Considerations

The Study does not consider the potential radiological consequences of the release of tritium during a LOCA. Tritium is contained in both the primary system coolant and the reactor core. As a result of a suggestion by the Environmental Protection Agency, this area will be studied.

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